UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DAVID GEISEN	)	Docket No. IA-05-052
	)	
	)	ASLBP No. 06-845-01-EA

Exhibits – Volume 1
Exhibits 1 – 70

Office of the Secretary

RAS C-142

This electronic text represents the Commission's current NRC Enforcement Policy. In a notice published in the *Federal Register* on March 16, 2005, the Commission announced its intent to use the NRC public Web site and the NRC's Agencywide Document Access and Management System (ADAMS) to communicate its "General Statement of Policy and Procedure for NRC Enforcement Actions—Enforcement Policy," to discontinue publication of the paper document, NUREG-1600, and to simplify the official policy statement title. The NRC is taking these actions because the policy statement is available electronically on the NRC public Web site and is widely known as the "NRC Enforcement Policy."

NRC ENFORCEMENT POLICY

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U.S. NRC
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NRC ENFORCEMENT POLICY

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Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)

Interim Enforcement Policy Regarding the Use of Alternative Dispute Resolution

PREFACE

The following policy statement describes the enforcement policy and procedures that the U.S. Nuclear Regulatory Commission (NRC or Commission) and its staff intends to follow in initiating and reviewing enforcement actions in response to violations of NRC requirements. This statement of general policy and procedure is publically available on the NRC public Web site and the NRC's Agencywide Document Access and Management System (ADAMS) to foster its widespread dissemination. However, this is a policy statement and not a regulation. The Commission may deviate from this statement of policy as appropriate under the circumstances of a particular case.

I. INTRODUCTION AND PURPOSE

The Atomic Energy Act of 1954, as amended, establishes "adequate protection" as the standard of safety on which NRC regulations are based. In the context of NRC regulations, safety means avoiding undue risk or, stated another way, providing reasonable assurance of adequate protection of workers and the public in connection with the use of source, byproduct and special nuclear materials.

While safety is the fundamental regulatory objective, compliance with NRC requirements plays an important role in giving the NRC confidence that safety is being maintained. NRC requirements, including technical specifications, other license conditions, orders, and regulations, have been designed to ensure adequate protection -- which corresponds to "no undue risk to public health and safety" -- through acceptable design, construction, operation, maintenance, modification, and quality assurance measures. In the context of risk-informed regulation, compliance plays a very important role in ensuring that key assumptions used in underlying risk and engineering analyses remain valid.

While adequate protection is presumptively assured by compliance with NRC requirements, circumstances may arise where new information reveals that an unforeseen hazard exists or that there is a substantially greater potential for a known hazard to occur. In such situations, the NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety.

The NRC also has the authority to exercise discretion to permit continued operations -- despite the existence of a noncompliance -- where the noncompliance is not significant from a risk perspective and does not, in the particular circumstances, pose an undue risk to public health and safety. When noncompliance occurs, the NRC must evaluate the degree of risk posed by that noncompliance to determine if specific immediate action is required. Where needed to ensure adequate protection of public health and safety, the NRC may demand immediate licensee action, up to and including a shutdown or cessation of licensed activities.

Based on the NRC's evaluation of noncompliance, the appropriate action could include refraining from taking any action, taking specific enforcement action, issuing orders, or providing input to other regulatory actions or assessments, such as increased oversight (e.g., increased inspection). Since some requirements are more important to safety than others, the NRC endeavors to use a risk-informed approach when applying NRC resources to the oversight of licensed activities, including enforcement activities.

The primary purpose of the NRC's Enforcement Policy is to support the NRC's overall safety mission in protecting the public health and safety and the environment. Consistent with that purpose, the policy endeavors to:

- Deter noncompliance by emphasizing the importance of compliance with NRC requirements, and

- Encourage prompt identification and prompt, comprehensive correction of violations of NRC requirements.

Therefore, licensees,[1] contractors,[2] and their employees who do not achieve the high standard of compliance which the NRC expects will be subject to enforcement sanctions. Each enforcement action is dependent on the circumstances of the case. However, in no case will licensees who cannot achieve and maintain adequate levels of safety be permitted to continue to conduct licensed activities.

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[1]This policy primarily addresses the activities of NRC licensees and applicants for NRC licenses. However, this policy provides for taking enforcement action against non-licensees and individuals in certain cases. These non-licensees include contractors and subcontractors, holders of, or applicants for, NRC approvals, e.g., certificates of compliance, early site permits, or standard design certificates, and the employees of these non-licensees. Specific guidance regarding enforcement action against individuals and non-licensees is addressed in Sections VIII and X, respectively.

[2]The term "contractor" as used in this policy includes vendors who supply products or services to be used in an NRC-licensed facility or activity.

II. STATUTORY AUTHORITY AND PROCEDURAL FRAMEWORK

A. Statutory Authority

The NRC's enforcement jurisdiction is drawn from the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act (ERA) of 1974, as amended.

Section 161 of the Atomic Energy Act authorizes the NRC to conduct inspections and investigations and to issue orders as may be necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property. Section 186 authorizes the NRC to revoke licenses under certain circumstances (e.g., for material false statements, in response to conditions that would have warranted refusal of a license on an original application, for a licensee's failure to build or operate a facility in accordance with the terms of the permit or license, and for violation of an NRC regulation). Section 234 authorizes the NRC to impose civil penalties not to exceed $100,000 per violation per day for the violation of certain specified licensing provisions of the Act, rules, orders, and license terms implementing these provisions, and for violations for which licenses can be revoked. In addition to the enumerated provisions in section 234, sections 84 and 147 authorize the imposition of civil penalties for violations of regulations implementing those provisions. Section 232 authorizes the NRC to seek injunctive or other equitable relief for violation of regulatory requirements.

Section 206 of the Energy Reorganization Act authorizes the NRC to impose civil penalties for knowing and conscious failures to provide certain safety information to the NRC.

Notwithstanding the $100,000 limit stated in the Atomic Energy Act, the Commission may impose higher civil penalties as provided by the Debt Collection Improvement Act of 1996. Under the Act, the Commission is required to modify civil monetary penalties to reflect inflation. The adjusted maximum civil penalty amount is reflected in 10 CFR 2.205 and this Policy Statement.

Chapter 18 of the Atomic Energy Act provides for varying levels of criminal penalties (i.e., monetary fines and imprisonment) for willful violations of the Act and regulations or orders issued under sections 65, 161(b), 161(i), or 161(o) of the Act. Section 223 provides that criminal penalties may be imposed on certain individuals employed by firms constructing or supplying basic components of any utilization facility if the individual knowingly and willfully violates NRC requirements such that a basic component could be significantly impaired. Section 235 provides that criminal penalties may be imposed on persons who interfere with inspectors. Section 236 provides that criminal penalties may be imposed on persons who attempt to or cause sabotage at a nuclear facility or to nuclear fuel. Alleged or suspected criminal violations of the Atomic Energy Act are referred to the Department of Justice for appropriate action.

B. Procedural Framework

Subpart B of 10 CFR Part 2 of NRC's regulations sets forth the procedures the NRC uses in exercising its enforcement authority. 10 CFR 2.201 sets forth the procedures for issuing Notices of Violation.

The procedure to be used in assessing civil penalties is set forth in 10 CFR 2.205. This regulation provides that the civil penalty process is initiated by issuing a Notice of Violation and Proposed Imposition of a Civil Penalty. The licensee or other person is provided an opportunity to contest the proposed imposition of a civil penalty in writing. After evaluation of the response, the civil penalty may be mitigated, remitted, or imposed. An opportunity is provided for a hearing if a civil penalty is imposed. If a civil penalty is not paid following a hearing or if a hearing is not requested, the matter may be referred to the U.S. Department of Justice to institute a civil action in District Court.

The procedure for issuing an order to institute a proceeding to modify, suspend, or revoke a license or to take other action against a licensee or other person subject to the jurisdiction of the Commission is set forth in 10 CFR 2.202. The licensee or any other person adversely affected by the order may request a hearing. The NRC is authorized to make orders immediately effective if required to protect the public health, safety, or interest, or if the violation is willful. Section 2.204 sets out the procedures for issuing a Demand for Information (Demand) to a licensee or other person subject to the Commission's jurisdiction for the purpose of determining whether an order or other enforcement action should be issued. The Demand does not provide hearing rights, as only information is being sought. A licensee must answer a Demand. An unlicensed person may answer a Demand by either providing the requested information or explaining why the Demand should not have been issued.

III. RESPONSIBILITIES

The Executive Director for Operations (EDO) and the principal enforcement officers of the NRC, the Deputy Executive Director for Reactor Programs (DEDR)and the Deputy Executive Director for Materials, Research and State Programs (DEDMRS) have been delegated the authority to approve or issue all escalated enforcement actions.[3] The DEDR is responsible to the EDO for NRC enforcement programs. The Office of Enforcement (OE) exercises oversight of and implements the NRC enforcement program. The Director, OE, acts for the Deputy Executive Director in enforcement matters in his absence or as delegated.

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[3]The term "escalated enforcement action" as used in this policy means a Notice of Violation or civil penalty for any Severity Level I, II, or III violation (or problem); a Notice of Violation associated with an inspection finding that the Significance Determination Process evaluates as having low to moderate, or greater, safety significance (i.e., white, yellow, or red); or any order based upon a violation.

Subject to the oversight and direction of OE, and with the approval of the Deputy Executive Director, where necessary, the regional offices normally issue Notices of Violation and proposed civil penalties. However, subject to the same oversight as the regional offices, the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Material Safety and Safeguards (NMSS) may also issue Notices of Violation and proposed civil penalties for certain activities. Enforcement orders are normally issued by the Deputy Executive Director or the Director, OE. However, orders may also be issued by the EDO, especially those involving the more significant matters. The Directors of NRR and NMSS have also been delegated authority to issue orders, but it is expected that normal use of this authority by NRR and NMSS will be confined to actions not associated with compliance issues. The Chief Financial Officer has been delegated the authority to issue orders where licensees violate Commission regulations by nonpayment of license and inspection fees.

In recognition that the regulation of nuclear activities in many cases does not lend itself to a mechanistic treatment, judgment and discretion must be exercised in determining the severity levels of the violations and the appropriate enforcement sanctions, including the decision to issue a Notice of Violation, or to propose or impose a civil penalty and the amount of this penalty, after considering the general principles of this statement of policy and the significance of the violations and the surrounding circumstances.

Unless Commission consultation or notification is required by this policy, the NRC staff may depart, where warranted in the public's interest, from this policy as provided in Section VII, "Exercise of Discretion."

The Commission will be provided written notification for the following situations:

(1) All enforcement actions involving civil penalties or orders;

(2) The first time that discretion is exercised for a plant that meets the criteria of Section VII.B.2;

(3) (Where appropriate, based on the uniqueness or significance of the issue) when discretion is exercised for violations that meet the criteria of Section VII.B.6; and

(4) All Notices of Enforcement Discretion (NOEDs) issued involving natural events, such as severe weather conditions.

The Commission will be consulted prior to taking action in the following situations (unless the urgency of the situation dictates immediate action):

(1) An action affecting a licensee's operation that requires balancing the public health and safety or common defense and security implications of not operating against the potential radiological or other hazards associated with continued operation (cases involving severe weather

or other natural phenomena may be addressed by the NRC staff without prior Commission consultation in accordance with Section VII.C);

(2) Proposals to impose a civil penalty for a single violation or problem that is greater than 3 times the Severity Level I value shown in Table 1A for that class of licensee;

(3) Any proposed enforcement action that involves a Severity Level I violation;

(4) Any action the EDO believes warrants Commission involvement;

(5) Any proposed enforcement case involving an Office of Investigations (OI) report where the NRC staff (other than the OI staff) does not arrive at the same conclusions as those in the OI report concerning issues of intent if the Director of OI concludes that Commission consultation is warranted; and

(6) Any proposed enforcement action on which the Commission asks to be consulted.

IV. SIGNIFICANCE OF VIOLATIONS

Regulatory requirements[4] have varying degrees of safety, safeguards, or environmental significance. Therefore, the relative importance or significance of each violation is assessed as the first step in the enforcement process.

A. Assessing Significance

In assessing the significance of a noncompliance, the NRC considers four specific issues: (1) actual safety consequences; (2) potential safety consequences, including the consideration of risk information; (3) potential for impacting the NRC's ability to perform its regulatory function; and (4) any willful aspects of the violation.

For certain types of violations at commercial nuclear power plants, the NRC relies on information from the Reactor Oversight Process's Significance Determination Process (SDP). The SDP is used to evaluate the actual and potential safety significance of inspection findings to provide a risk-informed framework for discussing and communicating the significance of inspection findings. Violations associated with findings evaluated through the SDP are addressed in Section IV.A.5. Violations at commercial nuclear power plants that are associated with inspection findings that *cannot* be evaluated through the SDP (i.e., violations that may impact the NRC's ability for oversight of licensed activities and violations that involve willfulness, including discrimination) are evaluated in accordance with the guidance in Sections IV.A.1 through IV.A.4 and Section IV.B. Violations that are associated with inspection

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[4]The term "requirement" as used in this policy means a legally binding requirement such as a statute, regulation, license condition, technical specification, or order.

findings with actual consequences are evaluated in accordance with the guidance in Section IV.A.5.c.

*1. **Actual Safety Consequences.*** In evaluating actual safety consequences, the NRC considers issues such as actual onsite or offsite releases of radiation, onsite or offsite radiation exposures, accidental criticalities, core damage, loss of significant safety barriers, loss of control of radioactive material or radiological emergencies. (See Section IV.A.5.c for guidance on violations that are associated with SDP findings with actual consequences.)

2. Potential Safety Consequences. In evaluating potential safety consequences, the NRC considers the *realistic* likelihood of affecting safety, i.e., the existence of credible scenarios with potentially significant actual consequences. The NRC will use risk information wherever possible in assessing significance and assigning severity levels. A higher severity may be warranted for violations that have greater risk significance and a lower severity level may be appropriate for issues that have low risk significance. Duration is an appropriate consideration in assessing the significance of violations.

3. Impacting the Regulatory Process. The NRC considers the safety implications of noncompliances that may impact the NRC's ability to carry out it statutory mission. Noncompliances may be significant because they may challenge the regulatory envelope upon which certain activities were licensed. These types of violations include failures such as: failures to provide complete and accurate information, failures to receive prior NRC approval for changes in licensed activities, failures to notify NRC of changes in licensed activities, failure to perform 10 CFR 50.59 analyses, reporting failures, etc., Even inadvertent reporting failures are important because many of the surveillance, quality control, and auditing systems on which both the NRC and its licensees rely in order to monitor compliance with safety standards are based primarily on complete, accurate, and timely recordkeeping and reporting. The existence of a regulatory process violation does not automatically mean that the issue is safety significant. In determining the significance of a violation, the NRC will consider appropriate factors for the particular regulatory process violation. These factors may include: the significance of the underlying issue, whether the failure actually impeded or influenced regulatory action, the level of individuals involved in the failure and the reasonableness of the failure given their position and training, and whether the failure invalidates the licensing basis. Factors to consider for failures to provide complete and accurate information are addressed in Section IX of this policy.

Unless otherwise categorized in the Supplements to this policy statement, the severity level of a violation involving the failure to make a required report to the NRC will be based upon the significance of and the circumstances surrounding the matter that should have been reported. However, the severity level of an untimely report, in contrast to no report, may be reduced depending on the circumstances surrounding the matter. A licensee will not normally be cited for a failure to report a condition or event unless the licensee was actually aware of the condition or event that it failed to report. A licensee will, on the other hand, normally be cited for a failure to report a condition or event if the licensee knew of the information to be reported, but did not recognize that it was required to make a report.

*4. **Willfulness.*** Willful violations are by definition of particular concern to the Commission because its regulatory program is based on licensees and their contractors, employees, and agents acting with integrity and communicating with candor. Willful violations cannot be tolerated by either the Commission or a licensee. Therefore, a violation may be considered more significant than the underlying noncompliance if it includes indications of willfulness. The term "willfulness" as used in this policy embraces a spectrum of violations ranging from deliberate intent to violate or falsify to and including careless disregard for requirements. Willfulness does not include acts which do not rise to the level of careless disregard, e.g., negligence or inadvertent clerical errors in a document submitted to the NRC. In determining the significance of a violation involving willfulness, consideration will be given to such factors as the position and responsibilities of the person involved in the violation (e.g., licensee official[5] or non-supervisory employee), the significance of any underlying violation, the intent of the violator (i.e., careless disregard or deliberateness), and the economic or other advantage, if any, gained as a result of the violation. The relative weight given to each of these factors in arriving at the significance assessment will be dependent on the circumstances of the violation. However, if a licensee refuses to correct a minor violation within a reasonable time such that it willfully continues, the violation should be considered at least more than minor. Licensees are expected to take significant remedial action in responding to willful violations commensurate with the circumstances such that it demonstrates the seriousness of the violation thereby creating a deterrent effect within the licensee's organization.

*5. **Significance Determination Process.*** The Reactor Oversight Process uses a Significance Determination Process (SDP) to determine the safety significance of most inspection findings identified at commercial nuclear power plants. Depending on their significance, inspection findings are assigned colors of green, white, yellow, or red. The Reactor Oversight Process uses an Agency Action Matrix to determine the appropriate agency response. If violations that are more than minor are associated with these inspection findings, they will be documented and may or may not be cited depending on the safety significance. These violations are not normally assigned severity levels, nor are they normally subject to civil penalties.

> NOTE: Violations associated with inspection findings that are *not evaluated through the SDP* will be assigned severity levels in accordance with Section IV.B and will be subject to civil penalties in accordance with Section VI.C.

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[5]The term "licensee official" as used in this policy statement means a first-line supervisor or above, a licensed individual, a radiation safety officer, or an authorized user of licensed material whether or not listed on a license. Notwithstanding an individual's job title, severity level categorization for willful acts involving individuals who can be considered licensee officials will consider several factors, including the position of the individual relative to the licensee's organizational structure and the individual's responsibilities relative to the oversight of licensed activities and to the use of licensed material.

a. Violations Associated with Findings of Very Low Safety Significance

Violations associated with findings that the SDP evaluates as having very low safety significance (i.e., green) will normally be described in inspection reports as Non-Cited Violations (NCVs). The finding will be categorized by the assessment process within the licensee response band. However, a Notice of Violation (NOV) will be issued if the issue meets one of the three applicable exceptions in Section VI.A.1. The Commission recognizes that violations exist below this category that are of minimal safety or environmental significance. While licensees must correct these minor violations, they don't normally warrant documentation in inspection reports and do not warrant enforcement action. To the extent such violations are described, they will be noted as violations of minor significance that are not subject to enforcement action.

b. Violations Associated with Findings of Low to Moderate, or Greater Safety Significance

Violations associated with findings that the SDP evaluates as having low to moderate safety significance (i.e., white), substantial safety significance (yellow), or high safety significance (red) will be cited in an NOV requiring a written response unless sufficient information is already on the docket. The finding will be assigned a color related to its significance for use by the assessment process. The Commission reserves the use of discretion for particularly significant violations (e.g. an accidental criticality) to assess civil penalties in accordance with Section 234 of the Atomic Energy Act of 1954, as amended.

c. Violations Associated with Actual Consequences

Violations that involve actual consequences such as an overexposure to the public or plant personnel above regulatory limits, failure to make the required notifications that impact the ability of Federal, State and local agencies to respond to an actual emergency preparedness (site area or general emergency), transportation event, or a substantial release of radioactive material, will be assigned severity levels and will be subject to civil penalties.

B. Assigning Severity Level

For purposes of determining the appropriate enforcement action, violations (except the majority of those associated with findings evaluated though the SDP) are normally categorized in terms of four levels of severity to show their relative importance or significance within each of the following eight activity areas:

I. Reactor Operations;
II. Facility Construction;
III. Safeguards;
IV. Health Physics;

V. Transportation;
VI. Fuel Cycle and Materials Operations;
VII. Miscellaneous Matters; and
VIII. Emergency Preparedness.

Licensed activities will be placed in the activity area most suitable in light of the particular violation involved, including activities not directly covered by one of the listed areas, e.g., export license activities. Within each activity area, Severity Level I has been assigned to violations that are the most significant and Severity Level IV violations are the least significant. Severity Level I and II violations are of very significant regulatory concern.[6] In general, violations that are included in these severity categories involve actual or high potential consequences on public health and safety. Severity Level III violations are cause for significant regulatory concern. Severity Level IV violations are less serious but are of more than minor concern. Violations at Severity Level IV involve noncompliance with NRC requirements that are not considered significant based on risk. This should not be misunderstood to imply that Severity Level IV issues have no risk significance.

The Commission recognizes that there are other violations of minor safety or environmental concern that are below the level of significance of Severity Level IV violations. While licensees must correct these minor violations, they don't normally warrant documentation in inspection reports or inspection records and do not warrant enforcement action. To the extent such violations are described, they will be noted as violations of minor significance that are not subject to enforcement action.

Comparisons of significance between activity areas are inappropriate. For example, the immediacy of any hazard to the public associated with Severity Level I violations in Reactor Operations is not directly comparable to that associated with Severity Level I violations in Facility Construction.

Supplements I through VIII provide examples and serve as guidance in determining the appropriate severity level for violations in each of the eight activity areas. However, the examples are neither exhaustive nor controlling. In addition, these examples do not create new requirements. Each is designed to illustrate the significance that the NRC places on a particular type of violation of NRC requirements. Each of the examples in the supplements is predicated on a violation of a regulatory requirement.

The NRC reviews each case being considered for enforcement action on its own merits to ensure that the severity of a violation is characterized at the level best suited to the significance of the particular violation.

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[6] Regulatory concern pertains to primary NRC regulatory responsibilities, *i.e.*, safety, safeguards, and the environment.

V. PREDECISIONAL ENFORCEMENT CONFERENCES

When the NRC learns of a potential violation for which escalated enforcement action appears to be warranted, or recurring nonconformance on the part of a contractor, the NRC may provide an opportunity for a predecisional enforcement conference with the licensee, contractor, or other person before taking enforcement action. The purpose of the predecisional enforcement conference is to obtain information that will assist the NRC in determining the appropriate enforcement action, such as: (1) a common understanding of facts, root causes, and missed opportunities associated with the apparent violations; (2) a common understanding of corrective actions taken or planned; and (3) a common understanding of the significance of issues and the need for lasting comprehensive corrective action.

The NRC may conduct Regulatory Conferences (in lieu of predecisional enforcement conferences) to discuss the significance of findings evaluated by the Reactor Oversight Process's SDP when apparent violations are associated with potentially significant findings. The purpose of Regulatory Conferences is to get information from licensees on the significance of findings evaluated through the SDP whether or not violations are involved. Because the significance assessment from the SDP determines whether or not escalated enforcement action will be issued (i.e., a Notice of Violation associated with a white, yellow, or red SDP finding), a subsequent predecisional enforcement conference is not normally necessary.

If the NRC concludes that it has sufficient information to make an informed enforcement decision involving a licensee, contractor, or vendor, a predecisional enforcement conference will not normally be held. If a predecisional enforcement conference is not held, the licensee may be given an opportunity to respond to a documented apparent violation (including its root causes and a description of planned or implemented corrective actions) before the NRC takes enforcement action. However, if the NRC has sufficient information to conclude that a civil penalty is not warranted, it may proceed to issue an enforcement action without first obtaining the licensee's response to the documented apparent violation.

The NRC will normally provide an opportunity for an individual to address apparent violations before the NRC takes escalated enforcement action. Whether an individual will be provided an opportunity for a predecisional enforcement conference or an opportunity to address an apparent violation in writing will depend on the circumstances of the case, including the severity of the issue, the significance of the action the NRC is contemplating, and whether the individual has already had an opportunity to address the issue (e.g., an Office of Investigation or a Department of Labor hearing).

During the predecisional enforcement conference, the licensee, contractor, or other persons will be given an opportunity to provide information consistent with the purpose of the conference, including an explanation to the NRC of the immediate corrective actions (if any) that were taken following identification of the potential violation or nonconformance and the long-term comprehensive actions that were taken or will be taken to prevent recurrence.

Licensees, contractors, or other persons will be told when a meeting is a predecisional enforcement conference.

A predecisional enforcement conference is a meeting between the NRC and the licensee. Conferences are normally held in the regional offices and are normally open to public observation. Predecisional enforcement conferences will not normally be open to the public if the enforcement action being contemplated:

(1) Would be taken against an individual, or if the action, though not taken against an individual, turns on whether an individual has committed wrongdoing;

(2) Involves significant personnel failures where the NRC has requested that the individual(s) involved be present at the conference;

(3) Is based on the findings of an NRC Office of Investigations report that has not been publicly disclosed; or

(4) Involves safeguards information, Privacy Act information, or information which could be considered proprietary;

In addition, conferences will not normally be open to the public if:

(5) The conference involves medical misadministrations or overexposures and the conference cannot be conducted without disclosing the exposed individual's name; or

(6) The conference will be conducted by telephone or the conference will be conducted at a relatively small licensee's facility.

Notwithstanding meeting any of these criteria, a predecisional enforcement conference may still be open if the conference involves issues related to an ongoing adjudicatory proceeding with one or more interveners or where the evidentiary basis for the conference is a matter of public record, such as an adjudicatory decision by the Department of Labor. In addition, notwithstanding the normal criteria for opening or closing predecisional enforcement conferences, conferences may either be open or closed to the public, with the approval of the Executive Director for Operations, after balancing the benefit of the public's observation against the potential impact on the agency's decision-making process in a particular case.

The NRC will notify the licensee that the predecisional enforcement conference will be open to public observation. Consistent with the agency's policy on open meetings (included on the NRC's Public Meeting Web site), the NRC intends to announce open conferences normally at least 10 calendar days in advance of conferences. Conferences will be announced on the Internet at the NRC Office of Enforcement's homepage (www.nrc.gov/OE) and on the Public Meeting Web site (www.nrc.gov/NRC/PUBLIC/meet.html). Individuals who do not have Internet access may get assistance on scheduled conferences by contacting the NRC staff at the Public Document Room, by calling toll-free 1-800-397-4209. In addition, the NRC will normally issue a press release and notify appropriate State liaison officers that a predecisional enforcement conference has been scheduled and that it is open to public observation.

The public attending open predecisional enforcement conferences may observe but may not participate in the conference. The purpose of conducting open conferences is not to maximize public attendance, but rather to provide the public with opportunities to be informed of NRC activities consistent with the NRC's ability to exercise its regulatory and safety responsibilities. Therefore, members of the public will be allowed access to the NRC regional offices to attend open enforcement conferences in accordance with the "Standard Operating Procedures For Providing Security Support For NRC Hearings and Meetings," published November 1, 1991 (56 FR 56251). These procedures provide that visitors may be subject to personnel screening, that signs, banners, posters, etc., not larger than 18" be permitted, and that disruptive persons may be removed. The open conference will be terminated if disruption interferes with a successful conference. NRC's Predecisional Enforcement Conferences (whether open or closed) normally will be held at the NRC's regional offices or in NRC Headquarters Offices and not in the vicinity of the licensee's facility.

For a case in which an NRC Office of Investigations (OI) report finds that discrimination as defined under 10 CFR 50.7 (or similar provisions in Parts 30, 40, 60, 70, or 72) has occurred, the OI report may be made public, subject to withholding certain information (i.e., after appropriate redaction), in which case the associated predecisional enforcement conference will normally be open to public observation. In a predecisional enforcement conference where a particular individual is being considered potentially responsible for the discrimination, the conference will remain closed. In either case (i.e., whether the conference is open or closed), the employee or former employee who was the subject of the alleged discrimination (hereafter referred to as "complainant") will normally be provided an opportunity to participate in the predecisional enforcement conference with the licensee/employer. This participation will normally be in the form of a complainant statement and comment on the licensee's presentation, followed in turn by an opportunity for the licensee to respond to the complainant's presentation. In cases where the complainant is unable to attend in person, arrangements will be made for the complainant's participation by telephone or an opportunity given for the complainant to submit a written response to the licensee's presentation. If the licensee chooses to forego an enforcement conference and, instead, responds to the NRC's findings in writing, the complainant will be provided the opportunity to submit written comments on the licensee's response. For cases involving potential discrimination by a contractor, any associated predecisional enforcement conference with the contractor would be handled similarly. These arrangements for complainant participation in the predecisional enforcement conference are not to be conducted or viewed in any respect as an adjudicatory hearing. The purpose of the complainant's participation is to provide information to the NRC to assist it in its enforcement deliberations.

A predecisional enforcement conference may not need to be held in cases where there is a full adjudicatory record before the Department of Labor. If a conference is held in such cases, generally the conference will focus on the licensee's corrective action. As with discrimination cases based on OI investigations, the complainant may be allowed to participate.

Members of the public attending open predecisional enforcement conferences will be reminded that (1) the apparent violations discussed at predecisional enforcement conferences are subject to further review and may be subject to change prior to any resulting enforcement action and (2) the statements of views or expressions of opinion made by NRC employees at predecisional enforcement conferences, or the lack thereof, are not intended to represent final determinations or beliefs.

When needed to protect the public health and safety or common defense and security, escalated enforcement action, such as the issuance of an immediately effective order, will be taken before the conference. In these cases, a conference may be held after the escalated enforcement action is taken.

VI. DISPOSITION OF VIOLATIONS

This section describes the various ways the NRC can disposition violations. The manner in which a violation is dispositioned is intended to reflect the seriousness of the violation and the circumstances involved. As previously stated, minor violations are not the subject of enforcement action. While licensees must correct these violations, they don't normally warrant documentation in inspection reports or inspection records. Other violations are documented and may be dispositioned as Non-Cited Violations, cited in Notices of Violation, or issued in conjunction with civil penalties or various types of orders. The NRC may also choose to exercise discretion and refrain from issuing enforcement action. (See Section VII.B, "Mitigation of Enforcement Sanctions.") As discussed further in Section VI.E, related administrative actions such as Notices of Nonconformance, Notices of Deviation, Confirmatory Action Letters, Letters of Reprimand, and Demands for Information are used to supplement the enforcement program. In determining the appropriate regulatory response, the NRC will consider enforcement actions taken by other Federal or State regulatory bodies having concurrent jurisdiction, such as in transportation matters.

A. Non-Cited Violation (NCV)

A Non-Cited Violation (NCV) is the term used to describe a method for dispositioning a Severity Level IV violation or a violation associated with a finding that the Reactor Oversight Process's SDP evaluates as having very low safety significance (i.e., green). These issues are documented as violations in inspection reports (or inspection records for some materials licensees) to establish public records of the violations, but are not cited in Notices of Violation which normally require written responses from licensees (see Section VI.B below). Dispositioning violations in this manner does not eliminate the NRC's emphasis on compliance with requirements nor the importance of maintaining safety. Licensees are still responsible for maintaining safety and compliance and must take steps to address corrective actions for these violations. While licensees are not required to provide written responses to NCVs, this approach allows licensees to dispute violations described as NCVs. The following sections describe the circumstances under which a violation may or may not be dispositioned as an NCV.

1. Power Reactor Licensees

Severity Level IV violations and violations associated with green SDP findings are normally dispositioned as NCVs. Violations dispositioned as NCVs will be described in inspection reports, although the NRC will close these violations based on their being entered into the licensee's corrective action program. At the time a violation is closed in an inspection report, the licensee may not have completed its corrective actions or begun the process to identify the root cause and develop action to prevent recurrence. Licensee actions will be taken commensurate with the established priorities and processes of the licensee's corrective action program. The NRC inspection program will provide an assessment of the effectiveness of the corrective action program. In addition to documentation in inspection reports, violations will be entered into the Plant Issues Matrix (PIM). Because the NRC will not normally obtain a written response from licensees describing actions taken to restore compliance and prevent recurrence of these violations, this enforcement approach places greater NRC reliance on licensee corrective action programs. Any one of the following circumstances will result in consideration of an NOV requiring a formal written response from a licensee.

a. The licensee failed to restore compliance within a reasonable time after a violation was identified.

b. The licensee did not place the violation into a corrective action program to address recurrence.

c. The violation is repetitive[7] as a result of inadequate corrective action, and was identified by the NRC. NOTE: This exception does not apply to violations associated with green SDP findings.

d. The violation was willful. Notwithstanding willfulness, an NCV may still be appropriate if:

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[7]A violation is considered "repetitive" if it could reasonably be expected to have been prevented by the licensee's corrective action for a previous violation or a previous licensee finding that occurred within the past 2 years of the inspection at issue, or the period within the last two inspections, whichever is longer.

(1) The licensee identified the violation and the information concerning the violation, if not required to be reported, was promptly provided to appropriate NRC personnel, such as a resident inspector or regional branch chief;

(2) The violation involved the acts of a low-level individual (and not a licensee official as defined in Section IV.A);

(3) The violation appears to be the isolated action of the employee without management involvement and the violation was not caused by lack of management oversight as evidenced by either a history of isolated willful violations or a lack of adequate audits or supervision of employees; and

(4) Significant remedial action commensurate with the circumstances was taken by the licensee such that it demonstrated the seriousness of the violation to other employees and contractors, thereby creating a deterrent effect within the licensee's organization.

The approval of the Director, Office of Enforcement, with consultation with the Deputy Executive Director as warranted, is required for dispositioning willful violations as NCVs.

2. - 7. [Reserved]

8. All Other Licensees

Severity Level IV violations that are dispositioned as NCVs will be described in inspection reports (or inspection records for some materials licensees) and will include a brief description of the corrective action the licensee has either taken or planned to take. Any one of the following circumstances will result in consideration of an NOV requiring a formal written response from a licensee.

a. The licensee failed to identify the violation;[8]

b. The licensee did not correct or commit to correct the violation within a reasonable time by specific corrective action committed to by the end of the inspection, including immediate corrective action and comprehensive corrective action to prevent recurrence; and

c. The violation is repetitive as a result of inadequate corrective action;

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[8]An NOV is warranted when a licensee identifies a violation as a result of an event where the root cause of the event is obvious or the licensee had prior opportunity to identify the problem but failed to take action that would have prevented the event. Disposition as an NCV may be warranted if the licensee demonstrated initiative in identifying the violation's root cause.

d. The violation was willful. Notwithstanding willfulness, an NCV may still be appropriate if it meets the criteria in Section VI.A.1.d.

The approval of the Director, Office of Enforcement, with consultation with the Deputy Executive Director as warranted, is required for dispositioning willful violations as NCVs.

B. Notice of Violation

A Notice of Violation is a written notice setting forth one or more violations of a legally binding requirement. The Notice of Violation normally requires the recipient to provide a written statement describing (1) the reasons for the violation or, if contested, the basis for disputing the violation; (2) corrective steps that have been taken and the results achieved; (3) corrective steps that will be taken to prevent recurrence; and (4) the date when full compliance will be achieved. The NRC may waive all or portions of a written response to the extent that relevant information has already been provided to the NRC in writing or documented in an NRC inspection report or inspection record. The NRC may require responses to Notices of Violation to be under oath. Normally, responses under oath will be required only in connection with Severity Level I, II, or III violations; violations associated with findings that the SDP evaluates as having low to moderate, or greater safety significance (i.e., white, yellow, or red); or orders.

Issuance of a Notice of Violation is normally the only enforcement action taken for Severity Level I, II, and III violations, except in cases where the criteria for issuance of civil penalties and orders, as set forth in Sections VI.C and VI.D, respectively, are met.

C. Civil Penalty

A civil penalty is a monetary penalty that may be imposed for violation of (1) certain specified licensing provisions of the Atomic Energy Act or supplementary NRC rules or orders; (2) any requirement for which a license may be revoked; or (3) reporting requirements under section 206 of the Energy Reorganization Act. Civil penalties are designed to deter future violations both by the involved licensee and other licensees conducting similar activities. Civil penalties also emphasize the need for licensees to identify violations and take prompt comprehensive corrective action.

Civil penalties are normally assessed for Severity Level I and II violations and knowing and conscious violations of the reporting requirements of section 206 of the Energy Reorganization Act. Civil penalties are considered for Severity Level III violations.

Civil penalties are also considered for violations associated with inspection findings evaluated through the Reactor Oversight Process's SDP that involved actual consequences, such as an overexposure to the public or plant personnel above regulatory limits, failure to make the required notifications that impact the ability of Federal, State and local agencies to respond to an actual emergency preparedness event (site area or general emergency), transportation event, or a substantial release of radioactive material. (Civil penalties are not proposed for violations associated with low to moderate, or greater safety significant findings absent actual consequences.)

Civil penalties are used to encourage prompt identification and prompt and comprehensive correction of violations, to emphasize compliance in a manner that deters future violations, and to serve to focus licensees' attention on significant violations.

Although management involvement, direct or indirect, in a violation may lead to an increase in the civil penalty, the lack of management involvement may not be used to mitigate a civil penalty. Allowing mitigation in the latter case could encourage the lack of management involvement in licensed activities and a decrease in protection of the public health and safety.

1. Base Civil Penalty

The NRC imposes different levels of penalties for different severity level violations and different classes of licensees, contractors, and other persons. Violations that involve loss, abandonment, or improper transfer or disposal of a sealed source or device are treated separately, regardless of the use or the type of licensee. Tables 1A and 1B show the base civil penalties for various reactor, fuel cycle, and materials programs, and for the loss, abandonment or improper transfer or disposal of a sealed source or device. (Civil penalties issued to individuals are determined on a case-by-case basis.) The structure of these tables generally takes into account the gravity of the violation as a primary consideration and the ability to pay as a secondary consideration. Generally, operations involving greater nuclear material inventories and greater potential consequences to the public and licensee employees receive higher civil penalties. Regarding the secondary factor of ability of various classes of licensees to pay the civil penalties, it is not the NRC's intention that the economic impact of a civil penalty be so severe that it puts a licensee out of business (orders, rather than civil penalties, are used when the intent is to suspend or terminate licensed activities) or adversely affects a licensee's ability to safely conduct licensed activities. The deterrent effect of civil penalties is best served when the amounts of the penalties take into account a licensee's ability to pay. In determining the amount of civil penalties for licensees for whom the tables do not reflect the ability to pay or the gravity of the violation, the NRC will consider necessary increases or decreases on a case-by-case basis. Normally, if a licensee can demonstrate financial hardship, the NRC will consider payments over time, including interest, rather than reducing the amount of the civil penalty. However, where a licensee claims financial hardship, the licensee will normally be required to address why it has sufficient resources to safely conduct licensed activities and pay license and inspection fees.

TABLE 1A--BASE CIVIL PENALTIES

a.	Power reactors and gaseous diffusion plants	$130,000
b.	Fuel fabricators authorized to possess Category I or II quantities of SNM	$65,000
c.	Fuel fabricators, industrial processors,[1] and independent spent fuel and monitored retrievable storage installations	$32,500
d.	Test reactors, mills and uranium conversion facilities, contractors, waste disposal licensees, industrial radiographers, and other large material users	$13,000
e.	Research reactors, academic, medical, or other small material users[2]	$6,500
f.	Loss, abandonment, or improper transfer or disposal of a sealed source or device, regardless of the use or type of licensee:[3]	
	1. Sources or devices with a total activity greater than 3.7×10^4 MBq (1 Curie), excluding hydrogen-3 (tritium)	$50,000
	2. Other sources or devices containing the materials and quantities listed in 10 CFR 31.5(c)(13)(i)	$16,500
	3. Sources and devices not otherwise described above	$6,500

[1]Large firms engaged in manufacturing or distribution of byproduct, source, or special nuclear material.

[2]This applies to nonprofit institutions not otherwise categorized in this table, mobile nuclear services, nuclear pharmacies, and physician offices.

[3]These base civil penalty amounts have been determined to be approximately three times the average cost of disposal. For specific cases, NRC may adjust these amounts to correspond to three times the actual expected cost of authorized disposal.

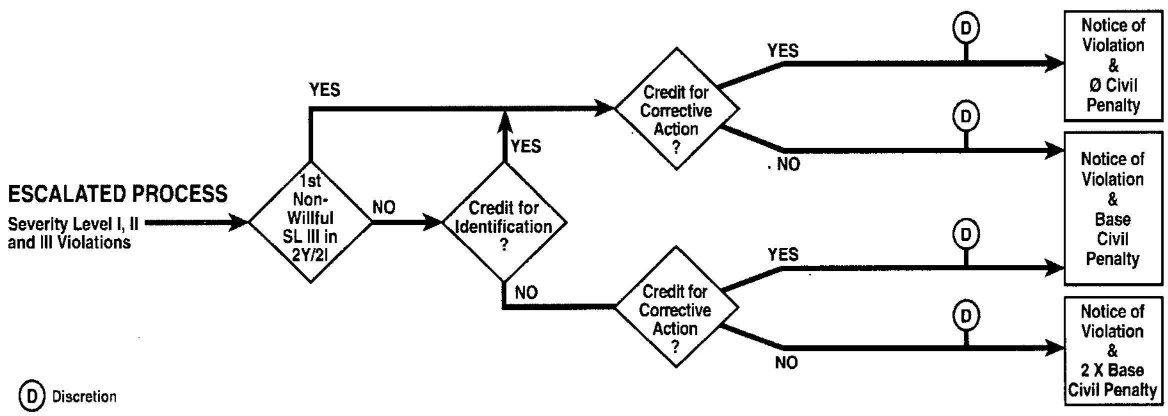
TABLE 1B--BASE CIVIL PENALTIES

Severity Level	Base Civil Penalty Amount (Percent of amount listed in Table 1A)
I	100%
II	80%
III	50%

2. Civil Penalty Assessment

In an effort to (1) emphasize the importance of adherence to requirements and (2) reinforce prompt self-identification of problems and root causes and prompt and comprehensive correction of violations, the NRC reviews each proposed civil penalty on its own merits and, after considering all relevant circumstances, may adjust the base civil penalties shown in Table 1A and 1B for Severity Level I, II, and III violations as described below.

The civil penalty assessment process considers four decisional points: (a) whether the licensee has had any previous escalated enforcement action (regardless of the activity area) during the past 2 years or past 2 inspections, whichever is longer; (b) whether the licensee should be given credit for actions related to identification; (c) whether the licensee's corrective actions are prompt and comprehensive; and (d) whether, in view of all the circumstances, the matter in question requires the exercise of discretion. Although each of these decisional points may have several associated considerations for any given case, the outcome of the assessment process for each violation or problem, absent the exercise of discretion, is limited to one of the following three results: no civil penalty, a base civil penalty, or a base civil penalty escalated by 100 percent. The flow chart presented below is a graphic representation of the civil penalty assessment process.



a. Initial Escalated Action

When the NRC determines that a non-willful Severity Level III violation or problem has occurred, and the licensee has not had any previous escalated actions (regardless of the activity area) during the past 2 years or 2 inspections, whichever is longer, the NRC will consider whether the licensee's corrective action for the present violation or problem is reasonably prompt and comprehensive (see the discussion under Section VI.C.2.c, below). Using 2 years as the basis for assessment is expected to cover most situations, but considering a slightly longer or shorter period might be warranted based on the circumstances of a particular case. The starting point of this period should be considered the date when the licensee was put on notice of the need to take corrective action. For a licensee-identified violation or an event, this would be when the licensee is aware that a problem or violation exists requiring corrective action. For an NRC-identified violation, the starting point would be when the NRC puts the licensee on notice, which could be during the inspection, at the inspection exit meeting, or as part of post-inspection communication.

If the corrective action is judged to be prompt and comprehensive, a Notice of Violation normally should be issued with no associated civil penalty. If the corrective action is judged to be less than prompt and comprehensive, the Notice of Violation normally should be issued with a base civil penalty.

b. Credit for Actions Related to Identification

(1) If a Severity Level I or II violation or a willful Severity Level III violation has occurred--or if, during the past 2 years or 2 inspections, whichever is longer, the licensee has been issued at least one other escalated action--the civil penalty assessment should normally consider the factor of identification in addition to corrective action (see the discussion under Section VI.C.2.c, below). In these circumstances, the NRC should consider whether the licensee should be given credit for actions related to identification.

In each case, the decision should be focused on identification of the problem requiring corrective action. In other words, although giving credit for *Identification* and *Corrective Action* should be separate decisions, the concept of *Identification* presumes that the identifier recognizes the existence of a problem, and understands that corrective action is needed. The decision on *Identification* requires considering all the circumstances of identification including:

(i) Whether the problem requiring corrective action was NRC-identified, licensee-identified, or revealed through an event[9];

(ii) Whether prior opportunities existed to identify the problem requiring corrective action, and if so, the age and number of those opportunities;

(iii) Whether the problem was revealed as the result of a licensee self-monitoring effort, such as conducting an audit, a test, a surveillance, a design review, or troubleshooting;

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[9]An "event," as used here, means (1) an event characterized by an active adverse impact on equipment or personnel, readily obvious by human observation or instrumentation, or (2) a radiological impact on personnel or the environment in excess of regulatory limits, such as an overexposure, a release of radioactive material above NRC limits, or a loss of radioactive material. For example, an equipment failure discovered through a spill of liquid, a loud noise, the failure to have a system respond properly, or an annunciator alarm would be considered an event; a system discovered to be inoperable through a document review would not. Similarly, if a licensee discovered, through quarterly dosimetry readings, that employees had been inadequately monitored for radiation, the issue would normally be considered licensee-identified; however, if the same dosimetry readings disclosed an overexposure, the issue would be considered an event.

(iv) For a problem revealed through an event, the ease of discovery, and the degree of licensee initiative in identifying the root cause of the problem and any associated violations;

(v) For NRC-identified issues, whether the licensee would likely have identified the issue in the same time-period if the NRC had not been involved;

(vi) For NRC-identified issues, whether the licensee should have identified the issue (and taken action) earlier; and

(vii) For cases in which the NRC identifies the overall problem requiring corrective action (e.g., a programmatic issue), the degree of licensee initiative or lack of initiative in identifying the problem or problems requiring corrective action.

(2) Although some cases may consider all of the above factors, the importance of each factor will vary based on the type of case as discussed in the following general guidance:

(i) **Licensee-Identified**. When a problem requiring corrective action is licensee-identified (i.e., identified before the problem has resulted in an event), the NRC should normally give the licensee credit for actions related to identification, regardless of whether prior opportunities existed to identify the problem.

(ii) **Identified Through an Event**. When a problem requiring corrective action is identified through an event, the decision on whether to give the licensee credit for actions related to identification normally should consider the ease of discovery, whether the event occurred as the result of a licensee self-monitoring effort (i.e., whether the licensee was "looking for the problem"), the degree of licensee initiative in identifying the problem or problems requiring corrective action, and whether prior opportunities existed to identify the problem.

Any of these considerations may be overriding if particularly noteworthy or particularly egregious. For example, if the event occurred as the result of conducting a surveillance or similar self-monitoring effort (i.e., the licensee was looking for the problem), the licensee should normally be given credit for identification. Even if the problem was easily discovered (e.g., revealed by a large spill of liquid), the NRC may choose to give credit because noteworthy licensee effort was exerted in ferreting out the root cause and associated violations, or simply because no prior opportunities (e.g., procedural cautions, post-maintenance testing, quality control failures, readily observable parameter trends, or repeated or locked-in annunciator warnings) existed to identify the problem.

(iii) **NRC-Identified**. When a problem requiring corrective action is NRC-identified, the decision on whether to give the licensee credit for actions related to *Identification* should normally be based on an additional question: should the licensee have reasonably identified the problem (and taken action) earlier?

In most cases, this reasoning may be based simply on the ease of the NRC inspector's discovery (e.g., conducting a walkdown, observing in the control room, performing a confirmatory NRC radiation survey, hearing a cavitating pump, or finding a valve obviously out of position). In some cases, the licensee's missed opportunities to identify the problem might include a similar previous violation, NRC or industry notices, internal audits, or readily observable trends.

If the NRC identifies the violation but concludes that, under the circumstances, the licensee's actions related to *Identification* were not unreasonable, the matter would be treated as licensee-identified for purposes of assessing the civil penalty. In such cases, the question of *Identification* credit shifts to whether the licensee should be penalized for NRC's identification of the problem.

(iv) **Mixed Identification**. For "mixed" identification situations (i.e., where multiple violations exist, some NRC-identified, some licensee-identified, or where the NRC prompted the licensee to take action that resulted in the identification of the violation), the NRC's evaluation should normally determine whether the licensee could reasonably have been expected to identify the violation in the NRC's absence. This determination should consider, among other things, the timing of the NRC's discovery, the information available to the licensee that caused the NRC concern, the specificity of the NRC's concern, the scope of the licensee's efforts, the level of licensee resources given to the investigation, and whether the NRC's path of analysis had been dismissed or was being pursued in parallel by the licensee.

In some cases, the licensee may have addressed the isolated symptoms of each violation (and may have identified the violations), but failed to recognize the common root cause and taken the necessary comprehensive action. Where this is true, the decision on whether to give licensee credit for actions related to *Identification* should focus on identification of *the problem requiring corrective action* (e.g., the programmatic breakdown). As such, depending on the chronology of the various violations, the earliest of the individual violations might be considered missed opportunities for the licensee to have identified the larger problem.

(v) **Missed Opportunities to Identify**. Missed opportunities include prior notifications or missed opportunities to identify or prevent violations such as (1) through normal surveillances, audits, or quality assurance (QA) activities; (2) through prior notice, i.e., specific NRC or industry notification; or (3) through other reasonable indication of a potential problem or violation, such as observations of employees and contractors, and failure to take effective corrective steps. It may include findings of the NRC, the licensee, or industry made at other facilities operated by the licensee where it is reasonable to expect the licensee to take action to identify or prevent similar problems at the facility subject to the enforcement action at issue. In assessing this factor, consideration will be given to, among other things, the opportunities available to discover the violation, the ease of discovery, the similarity between the violation and the notification, the period of time between when the violation occurred and when the notification was issued, the action taken (or planned) by the licensee in response to the notification, and the level of management review that the notification received (or should have received).

The evaluation of missed opportunities should normally depend on whether the information available to the licensee should reasonably have caused action that would have prevented the violation. Missed opportunities is normally not applied where the licensee appropriately reviewed the opportunity for application to its activities and reasonable action was either taken or planned to be taken within a reasonable time.

In some situations the missed opportunity is a violation in itself. In these cases, unless the missed opportunity is a Severity Level III violation in itself, the missed opportunity violation may be grouped with the other violations into a single Severity Level III "problem." However, if the missed opportunity is the *only* violation, then it should not normally be counted twice (i.e., both as the violation and as a missed opportunity--"double counting") unless the number of opportunities missed was particularly significant.

The timing of the missed opportunity should also be considered. While a rigid time-frame is unnecessary, a 2-year period should generally be considered for consistency in implementation, as the period reflecting relatively current performance.

(3) When the NRC determines that the licensee should receive credit for actions related to *Identification*, the civil penalty assessment should normally result in either no civil penalty or a base civil penalty, based on whether *Corrective Action* is judged to be reasonably prompt and comprehensive. When the licensee is *not* given credit for actions related to *Identification*, the civil penalty assessment should normally result in a Notice of Violation with either a base civil penalty or a base civil penalty escalated by 100 percent, depending on the quality of *Corrective Action*, because the licensee's performance is clearly not acceptable.

c. Credit for Prompt and Comprehensive Corrective Action

The purpose of the *Corrective Action* factor is to encourage licensees to (1) take the immediate actions necessary upon discovery of a violation that will restore safety and compliance with the license, regulation(s), or other requirement(s); and (2) develop and implement (in a timely manner) the lasting actions that will not only prevent recurrence of the violation at issue, but will be appropriately comprehensive, given the significance and complexity of the violation, to prevent occurrence of violations with similar root causes.

Regardless of other circumstances (e.g., past enforcement history, identification), the licensee's corrective actions should always be evaluated as part of the civil penalty assessment process. As a reflection of the importance given to this factor, an NRC judgment that the licensee's corrective action has not been prompt and comprehensive will always result in issuing at least a base civil penalty.

In assessing this factor, consideration will be given to the timeliness of the corrective action (including the promptness in developing the schedule for long term corrective action), the adequacy of the licensee's root cause analysis for the violation, and, given the significance and complexity of the issue, the comprehensiveness of the corrective action (i.e., whether the action is focused narrowly to the specific violation or broadly to the general area of concern). Even in cases when the NRC, at the time of the enforcement conference, identifies additional peripheral or minor corrective action still to be taken, the licensee may be given credit in this area, as long as the licensee's actions addressed the underlying root cause and are considered sufficient to prevent recurrence of the violation and similar violations.

Normally, the judgment of the adequacy of corrective actions will hinge on whether the NRC had to take action to focus the licensee's evaluative and corrective process in order to obtain comprehensive corrective action. This will normally be judged at the time of the predecisional enforcement conference (e.g., by outlining substantive additional areas where corrective action is needed). Earlier informal discussions between the licensee and NRC inspectors or management may result in improved corrective action, but should not normally be a basis to deny credit for *Corrective Action.* For cases in which the licensee does not get credit for actions related to *Identification* because the NRC identified the problem, the assessment of the licensee's corrective action should begin from the time when the NRC put the licensee on notice of the problem. Notwithstanding eventual good comprehensive corrective action, if immediate corrective action was not taken to restore safety and compliance once the violation was identified, corrective action would not be considered prompt and comprehensive.

Corrective action for violations involving discrimination should normally only be considered comprehensive if the licensee takes prompt, comprehensive corrective action that (1) addresses the broader environment for raising safety concerns in the workplace, and (2) provides a remedy for the particular discrimination at issue.

In response to violations of 10 CFR 50.59, corrective action should normally be considered prompt and comprehensive only if the licensee --

(i) Makes a prompt decision on operability; and either

(ii) Makes a prompt evaluation under 10 CFR 50.59 if the licensee intends to maintain the facility or procedure in the as found condition; or

(iii) Promptly initiates corrective action consistent with Criterion XVI of 10 CFR 50, Appendix B, if it intends to restore the facility or procedure to the FSAR description.

d. Exercise of Discretion

As provided in Section VII, "Exercise of Discretion," discretion may be exercised by either escalating or mitigating the amount of the civil penalty determined after applying the civil penalty adjustment factors to ensure that the proposed civil penalty reflects all relevant circumstances of the particular case. However, in no instance will a civil penalty for any one violation exceed $130,000 per day.

D. Orders

An order is a written NRC directive to modify, suspend, or revoke a license; to cease and desist from a given practice or activity; or to take such other action as may be proper (see 10 CFR 2.202). Orders may also be issued in lieu of, or in addition to, civil penalties, as appropriate for Severity Level I, II, or III violations. Orders may be issued as follows:

1. License Modification orders are issued when some change in licensee equipment, procedures, personnel, or management controls is necessary.

2. Suspension Orders may be used:

(a) To remove a threat to the public health and safety, common defense and security, or the environment;

(b) To stop facility construction when,

(i) Further work could preclude or significantly hinder the identification or correction of an improperly constructed safety-related system or component; or

(ii) The licensee's quality assurance program implementation is not adequate to provide confidence that construction activities are being properly carried out;

(c) When the licensee has not responded adequately to other enforcement action;

(d) When the licensee interferes with the conduct of an inspection or investigation; or

(e) For any reason not mentioned above for which license revocation is legally authorized.

Suspensions may apply to all or part of the licensed activity. Ordinarily, a licensed activity is not suspended (nor is a suspension prolonged) for failure to comply with requirements where such failure is not willful and adequate corrective action has been taken.

3. Revocation Orders may be used:

(a) When a licensee is unable or unwilling to comply with NRC requirements;

(b) When a licensee refuses to correct a violation;

(c) When licensee does not respond to a Notice of Violation where a response was required;

(d) When a licensee refuses to pay an applicable fee under the Commission's regulations; or

(e) For any other reason for which revocation is authorized under section 186 of the Atomic Energy Act (e.g., any condition which would warrant refusal of a license on an original application).

4. Cease and Desist Orders may be used to stop an unauthorized activity that has continued after notification by the NRC that the activity is unauthorized.

5. Orders to non-licensees, including contractors and subcontractors, holders of NRC approvals, e.g., certificates of compliance, early site permits, standard design certificates, or applicants for any of them, and to employees of any of the foregoing, are used when the NRC has identified deliberate misconduct that may cause a licensee to be in violation of an NRC requirement or where incomplete or inaccurate information is deliberately submitted or where the NRC loses its reasonable assurance that the licensee will meet NRC requirements with that person involved in licensed activities.

Unless a separate response is warranted under 10 CFR 2.201, a Notice of Violation need not be issued where an order is based on violations described in the order. The violations described in an order need not be categorized by severity level.

Orders are made effective immediately, without prior opportunity for hearing, whenever it is determined that the public health, interest, or safety so requires, or when the order is responding to a violation involving willfulness. Otherwise, a prior opportunity for a hearing on the order is afforded. For cases in which the NRC believes a basis could reasonably exist for not taking the action as proposed, the licensee will ordinarily be afforded an opportunity to show why the order should not be issued in the proposed manner by way of a Demand for Information. (See 10 CFR 2.204)

E. Related Administrative Actions

In addition to NCVs, NOVs, civil penalties, and orders, the NRC also uses administrative actions, such as Notices of Deviation, Notices of Nonconformance, Confirmatory Action Letters, Letters of Reprimand, and Demands for Information to supplement its enforcement program. The NRC expects licensees and contractors to adhere to any obligations and commitments resulting from these actions and will not hesitate to issue appropriate orders to ensure that these obligations and commitments are met.

1. **Notices of Deviation** are written notices describing a licensee's failure to satisfy a commitment where the commitment involved has not been made a legally binding requirement. A Notice of Deviation requests that a licensee provide a written explanation or statement describing corrective steps taken (or planned), the results achieved, and the date when corrective action will be completed.

2. **Notices of Nonconformance** are written notices describing contractors' failures to meet commitments which have not been made legally binding requirements by NRC. An example is a commitment made in a procurement contract with a licensee as required by 10 CFR Part 50, Appendix B. Notices of Nonconformances request that non-licensees provide written explanations or statements describing corrective steps (taken or planned), the results achieved, the dates when corrective actions will be completed, and measures taken to preclude recurrence.

3. **Confirmatory Action Letters** are letters confirming a licensee's or contractor's agreement to take certain actions to remove significant concerns about health and safety, safeguards, or the environment.

4. **Letters of Reprimand** are letters addressed to individuals subject to Commission jurisdiction identifying a significant deficiency in their performance of licensed activities.

5. **Demands for Information** are demands for information from licensees or other persons for the purpose of enabling the NRC to determine whether an order or other enforcement action should be issued.

VII. EXERCISE OF DISCRETION

Notwithstanding the normal guidance contained in this policy, as provided in Section III, "Responsibilities," the NRC may choose to exercise discretion and either escalate or mitigate enforcement sanctions within the Commission's statutory authority to ensure that the resulting enforcement action takes into consideration all of the relevant circumstances of the particular case.

A. Escalation of Enforcement Sanctions

The NRC considers violations categorized at Severity Level I, II, or III to be of significant regulatory concern. The NRC also considers violations associated with findings that the Reactor Oversight Process's Significance Determination Process evaluates as having low to moderate, or greater safety significance (i.e., white, yellow, or red) to be of significant regulatory concern. If the application of the normal guidance in this policy does not result in an appropriate sanction, with the approval of the Deputy Executive Director and consultation with the EDO and Commission, as warranted, the NRC may apply its full enforcement authority where the action is warranted. NRC action may include (1) escalating civil penalties; (2) issuing appropriate orders; and (3) assessing civil penalties for continuing violations on a per day basis, up to the statutory limit of $130,000 per violation, per day.

1. Civil Penalties

Notwithstanding the outcome of the normal civil penalty assessment process addressed in Section VI.C, the NRC may exercise discretion by either proposing a civil penalty where application of the factors would otherwise result in zero penalty or by escalating the amount of the resulting civil penalty (i.e., base or twice the base civil penalty) to ensure that the proposed civil penalty reflects the significance of the circumstances. The Commission will be notified if the deviation in the amount of the civil penalty proposed under this discretion from the amount of the civil penalty assessed under the normal process is more than two times the base civil penalty shown in Tables 1A and 1B. Examples when this discretion should be considered include, but are not limited to the following:

(a) Problems categorized at Severity Level I or II;

(b) Overexposures, or releases of radiological material in excess of NRC requirements;

(c) Situations involving particularly poor licensee performance, or involving willfulness;

(d) Situations when the licensee's previous enforcement history has been particularly poor, or when the current violation is directly repetitive of an earlier violation;

(e) Situations when the violation results in a substantial increase in risk, including cases in which the duration of the violation has contributed to the substantial increase;

(f) Situations when the licensee made a conscious decision to be in noncompliance in order to obtain an economic benefit;

(g) Cases involving the loss, abandonment, or improper transfer or disposal of a sealed source or device. Notwithstanding the outcome of the normal civil penalty assessment process, these cases normally should result in a civil penalty of at least the base amount; or

(h) Severity Level II or III violations associated with departures from the Final Safety Analysis Report identified after March 30, 2000, for risk-significant items as defined by the licensee's maintenance rule program and March 30, 2001, for all other issues. Such a violation or problem would consider the number and nature of the violations, the severity of the violations, whether the violations were continuing, and who identified the violations (and if the licensee identified the violation, whether exercise of Section VII.B.3 enforcement discretion is warranted.)

2. Orders

The NRC may, where necessary or desirable, issues orders in conjunction with or in lieu of civil penalties to achieve or formalize corrective actions and to deter further recurrence of serious violations.

3. Daily Civil Penalties

In order to recognize the added significance for those cases where a very strong message is warranted for a significant violation that continues for more than one day, the NRC may exercise discretion and assess a separate violation and attendant civil penalty up to the statutory limit of $130,000 for each day the violation continues. The NRC may exercise this discretion if a licensee was aware of or clearly should have been aware of a violation, or if the licensee had an opportunity to identify and correct the violation but failed to do so.

B. Mitigation of Enforcement Sanctions

The NRC may exercise discretion and refrain from issuing a civil penalty and/or a Notice of Violation after considering the general principles of this statement of policy and the surrounding circumstances.[10] The approval of the Director, Office of Enforcement, in consultation with the Deputy Executive Director, as warranted, is required for exercising discretion of the type described in Sections VII.B.2 through VII.B.6. The circumstances under which mitigation discretion should be considered include, but are not limited to the following:

1. [Reserved]

[10] The mitigation discretion described in Sections VII.B.2 - VII.B.6 does not normally apply to violations associated with issues evaluated by the SDP. The Reactor Oversight Process will use the Agency Action Matrix to determine the agency response to performance issues. The Agency Action Matrix has provisions to consider extenuating circumstances that were previously addressed through enforcement mitigation.

2. *Violations Identified During Extended Shutdowns or Work Stoppages*

The NRC may refrain from issuing a Notice of Violation or a proposed civil penalty for a Severity Level II, III, or IV violation that is identified after (i) the NRC has taken significant enforcement action based upon a major safety event contributing to an extended shutdown of an operating reactor or a material licensee (or a work stoppage at a construction site), or (ii) the licensee enters an extended shutdown or work stoppage related to generally poor performance over a long period of time, provided that the violation is documented in an inspection report (or inspection records for some material cases) and that it meets all of the following criteria:

(a) It was either licensee-identified as a result of a comprehensive program for problem identification and correction that was developed in response to the shutdown or identified as a result of an employee allegation to the licensee; (If the NRC identifies the violation and all of the other criteria are met, the NRC should determine whether enforcement action is necessary to achieve remedial action, or if discretion may still be appropriate.)

(b) It is based upon activities of the licensee prior to the events leading to the shutdown;

(c) It would not be categorized at Severity Level I;

(d) It was not willful; and

(e) The licensee's decision to restart the plant requires NRC concurrence.

3. *Violations Involving Old Design Issues*

The NRC may refrain from proposing a civil penalty for a Severity Level II or III violation involving a past problem, such as in engineering, design, or installation, if the violation is documented in an inspection report (or inspection records for some material cases) that includes a description of the corrective action and that it meets all of the following criteria:

(a) It was a licensee-identified as a result of its voluntary initiative;

(b) It was or will be corrected, including immediate corrective action and long term comprehensive corrective action to prevent recurrence, within a reasonable time following identification (this action should involve expanding the initiative, as necessary, to identify other failures caused by similar root causes); and

(c) It was not likely to be identified (after the violation occurred) by routine licensee efforts such as normal surveillance or quality assurance (QA) activities.

In addition, the NRC may refrain from issuing a Notice of Violation for a Severity Level II, III, or IV violation that meets the above criteria provided the violation was caused by conduct that is not reasonably linked to present performance (normally, violations that are at least 3 years old or violations occurring during plant construction) and there had not been prior notice so that the licensee should have reasonably identified the violation earlier. This exercise of discretion is to place a premium on licensees initiating efforts to identify and correct subtle violations that are not likely to be identified by routine efforts before degraded safety systems are called upon to work.

Section VII.B.3 discretion would not normally be applied to departures from the FSAR if:

(a) The NRC identifies the violation, unless it was likely in the NRC staff's view that the licensee would have identified the violation in light of the defined scope, thoroughness, and schedule of the licensee's initiative provided the schedule provides for completion of the licensee's initiative by March 30, 2000, for risk-significant items as defined by the licensee's maintenance rule program and by March 30, 2001, for all other issues;

(b) The licensee identifies the violation as a result of an event or surveillance or other required testing where required corrective action identifies the FSAR issue;

(c) The licensee identifies the violation but had prior opportunities to do so (was aware of the departure from the FSAR) and failed to correct it earlier;

(d) There is willfulness associated with the violation;

(e) The licensee fails to make a report required by the identification of the departure from the FSAR; or

(f) The licensee either fails to take comprehensive corrective action or fails to appropriately expand the corrective action program. The corrective action should be broad with a defined scope and schedule.

4. *Violations Identified Due to Previous Enforcement Action*

The NRC may refrain from issuing a Notice of Violation or a proposed civil penalty for a Severity Level II, III, or IV violation that is identified after the NRC has taken enforcement action, if the violation is documented in an inspection report (or inspection records for some material cases) that includes a description of the corrective action and that it meets all of the following criteria:

(a) It was licensee-identified as part of the corrective action for the previous enforcement action;

(b) It has the same or similar root cause as the violation for which enforcement action was issued;

(c) It does not substantially change the safety significance or the character of the regulatory concern arising out of the initial violation; and

(d) It was or will be corrected, including immediate corrective action and long term comprehensive corrective action to prevent recurrence, within a reasonable time following identification.

(e) It would not be categorized at Severity Level I;

5. *Violations Involving Certain Discrimination Issues*

Enforcement discretion may be exercised for discrimination cases when a licensee who, without the need for government intervention, identifies an issue of discrimination and takes prompt, comprehensive, and effective corrective action to address both the particular situation and the overall work environment for raising safety concerns. Similarly, enforcement may not be warranted where a complaint is filed with the Department of Labor (DOL) under Section 211 of the Energy Reorganization Act of 1974, as amended, but the licensee settles the matter before the DOL makes an initial finding of discrimination and addresses the overall work environment. Alternatively, if a finding of discrimination is made, the licensee may choose to settle the case before the evidentiary hearing begins. In such cases, the NRC may exercise its discretion not to take enforcement action when the licensee has addressed the overall work environment for raising safety concerns and has publicized that a complaint of discrimination for engaging in protected activity was made to the DOL, that the matter was settled to the satisfaction of the employee (the terms of the specific settlement agreement need not be posted), and that, if the DOL Area Office found discrimination, the licensee has taken action to positively reemphasize that discrimination will not be tolerated. Similarly, the NRC may refrain from taking enforcement action if a licensee settles a matter promptly after a person comes to the NRC without going to the DOL. Such discretion would normally not be exercised in cases in which the licensee does not appropriately address the overall work environment (*e.g.*, by using training, postings, revised policies or procedures, any necessary disciplinary action, etc., to communicate its policy against discrimination) or in cases that involve: allegations of discrimination as a result of providing information directly to the NRC, allegations of discrimination caused by a manager above first-line supervisor (consistent with current Enforcement Policy classification of Severity Level I or II violations), allegations of discrimination where a history of findings of discrimination (by the DOL or the NRC) or settlements suggests a programmatic rather than an isolated discrimination problem, or allegations of discrimination which appear particularly blatant or egregious.

6. *Violations Involving Special Circumstances*

Notwithstanding the outcome of the normal enforcement process addressed in Section VI.B or the normal civil penalty assessment process addressed in Section VI.C, the NRC may reduce or refrain from issuing a civil penalty or a Notice of Violation for a Severity Level II, III, or IV violation based on the merits of the case after considering the guidance in this statement of policy and such factors as the age of the violation, the significance of the violation, the clarity of the requirement, the appropriateness of the requirement, the overall sustained performance of the licensee has been particularly good, and other relevant circumstances, including any that may have changed since the violation. This discretion is expected to be exercised only where application of the normal guidance in the policy is unwarranted. In addition, the NRC may refrain from issuing enforcement action for violations resulting from matters not within a licensee's control, such as equipment failures that were not avoidable by reasonable licensee quality assurance measures or management controls. Generally, however, licensees are held responsible for the acts of their employees and contractors. Accordingly, this policy should not be construed to excuse personnel or contractor errors.

C. Notice of Enforcement Discretion for Power Reactors and Gaseous Diffusion Plants

On occasion, circumstances may arise where a power reactor's compliance with a Technical Specification (TS) Limiting Condition for Operation or with other license conditions would involve an unnecessary plant transient or performance of testing, inspection, or system realignment that is inappropriate with the specific plant conditions, or unnecessary delays in plant startup without a corresponding health and safety benefit. Similarly, for a gaseous diffusion plant (GDP), circumstances may arise where compliance with a Technical Safety Requirement (TSR) or technical specification or other certificate condition would unnecessarily call for a total plant shutdown or, notwithstanding that a safety, safeguards, or security feature was degraded or inoperable, compliance would unnecessarily place the plant in a transient or condition where those features could be required.

In these circumstances, the NRC staff may choose not to enforce the applicable TS, TSR, or other license or certificate condition. This enforcement discretion, designated as a Notice of Enforcement Discretion (NOED), will only be exercised if the NRC staff is clearly satisfied that the action is consistent with protecting the public health and safety. The NRC staff may also grant enforcement discretion in cases involving severe weather or other natural phenomena, based upon balancing the public health and safety or common defense and security of not operating against the potential radiological or other hazards associated with continued operation, and a determination that safety will not be impacted unacceptably by exercising this discretion. The Commission is to be informed expeditiously following the granting of a NOED in these situations. A licensee or certificate holder seeking the issuance of a NOED must provide a written justification, or in circumstances where good cause is shown, oral justification followed as soon as possible by written justification, that documents the safety basis for the request and provides whatever other information necessary for the NRC staff to make a decision on whether to issue a NOED.

For power reactors, the appropriate Regional Administrator, or his or her designee, may issue a NOED after consultation with the Director, Office of Nuclear Reactor Regulation, or his or her designee, to determine the appropriateness of granting a NOED where (1) the noncompliance is temporary and nonrecurring when an amendment is not practical or (2) if the expected noncompliance will occur during the brief period of time it requires the NRC staff to process an emergency or exigent license amendment under the provisions of 10 CFR 50.91 (a)(5) or (6). For gaseous diffusion plants, the appropriate Regional Administrator, or his or her designee, may issue and document a NOED where the noncompliance is temporary and nonrecurring and when an amendment is not practical. The Director, Office of Nuclear Materials Safety and Safeguards, or his or her designee, may issue a NOED if the expected noncompliance will occur during the brief period of time it requires the NRC staff to process a certificate amendment under 10 CFR 76.45. The person exercising enforcement discretion will document the decision.

For an operating reactor, this exercise of enforcement discretion is intended to minimize the potential safety consequences of unnecessary plant transients with the accompanying operational risks and impacts or to eliminate testing, inspection, or system realignment which is inappropriate for the particular plant conditions. For plants in a shutdown condition, exercising enforcement discretion is intended to reduce shutdown risk by, again, avoiding testing, inspection or system realignment which is inappropriate for the particular plant conditions, in that, it does not provide a safety benefit or may, in fact, be detrimental to safety in the particular plant condition. Exercising enforcement discretion for plants attempting to startup is less likely than exercising it for an operating plant, as simply delaying startup does not usually leave the plant in a condition in which it could experience undesirable transients. In such cases, the Commission would expect that discretion would be exercised with respect to equipment or systems only when it has at least concluded that, notwithstanding the conditions of the license: (1) the equipment or system does not perform a safety function in the mode in which operation is to occur; (2) the safety function performed by the equipment or system is of only marginal safety benefit, provided remaining in the current mode increases the likelihood of an unnecessary plant transient; or (3) the TS or other license condition requires a test, inspection, or system realignment that is inappropriate for the particular plant conditions, in that it does not provide a safety benefit, or may, in fact, be detrimental to safety in the particular plant condition.

For GDPs, the exercise of enforcement discretion would be used where compliance with a certificate condition would involve an unnecessary plant shutdown or, notwithstanding that a safety, safeguards, or security feature was degraded or inoperable, compliance would unnecessarily place the plant in a transient or condition where those features could be required. Such regulatory flexibility is needed because a total plant shutdown is not necessarily the best response to a plant condition. GDPs are designed to operate continuously and have never been shut down. Although portions can be shut down for maintenance, the NRC staff has been informed by the certificate holder that restart from a total plant shutdown may not be practical and the staff agrees that the design of a GDP does not make restart practical. Hence, the decision to place either GDP in plant-wide shutdown condition would be made only after determining that there is inadequate safety, safeguards, or security and considering the total impact of the shutdown on safety, the environment, safeguards, and security. A NOED would not be used for noncompliances with other than certificate requirements, or for situations where the certificate holder cannot demonstrate adequate safety, safeguards, or security.

The decision to exercise enforcement discretion does not change the fact that a violation will occur nor does it imply that enforcement discretion is being exercised for any violation that may have led to the violation at issue. In each case where the NRC staff has chosen to issue a NOED, enforcement action will normally be taken for the root causes, to the extent violations were involved, that led to the noncompliance for which enforcement discretion was used. The enforcement action is intended to emphasize that licensees and certificate holders should not rely on the NRC's authority to exercise enforcement discretion as a routine substitute for compliance or for requesting a license or certificate amendment.

Finally, it is expected that the NRC staff will exercise enforcement discretion in this area infrequently. Although a plant must shut down, refueling activities may be suspended, or plant startup may be delayed, absent the exercise of enforcement discretion, the NRC staff is under no obligation to take such a step merely because it has been requested. The decision to forego enforcement is discretionary. When enforcement discretion is to be exercised, it is to be exercised only if the NRC staff is clearly satisfied that the action is warranted from a health and safety perspective.

VIII. ENFORCEMENT ACTIONS INVOLVING INDIVIDUALS

Enforcement actions involving individuals, including licensed operators, are significant personnel actions, which will be closely controlled and judiciously applied. An enforcement action involving an individual will normally be taken only when the NRC is satisfied that the individual fully understood, or should have understood, his or her responsibility; knew, or should have known, the required actions; and knowingly, or with careless disregard (i.e., with more than mere negligence) failed to take required actions which have actual or potential safety significance. Most transgressions of individuals at the level of Severity Level III or IV violations will be handled by citing only the facility licensee.

More serious violations, including those involving the integrity of an individual (e.g., lying to the NRC) concerning matters within the scope of the individual's responsibilities, will be considered for enforcement action against the individual as well as against the facility licensee. However, action against the individual will not be taken if the improper action by the individual was caused by management failures. The following examples of situations illustrate this concept:

- Inadvertent individual mistakes resulting from inadequate training or guidance provided by the facility licensee.

- Inadvertently missing an insignificant procedural requirement when the action is routine, fairly uncomplicated, and there is no unusual circumstance indicating that the procedures should be referred to and followed step-by-step.

- Compliance with an express direction of management, such as the Shift Supervisor or Plant Manager, resulted in a violation unless the individual did not express his or her concern or objection to the direction.

- Individual error directly resulting from following the technical advice of an expert unless the advise was clearly unreasonable and the licensed individual should have recognized it as such.

- Violations resulting from inadequate procedures unless the individual used a faulty procedure knowing it was faulty and had not attempted to get the procedure corrected.

Listed below are examples of situations which could result in enforcement actions involving individuals, licensed or unlicensed. If the actions described in these examples are taken by a licensed operator or taken deliberately by an unlicensed individual, enforcement action may be taken directly against the individual. However, violations involving willful conduct not amounting to deliberate action by an unlicensed individual in these situations may result in enforcement action against a licensee that may impact an individual. The situations include, but are not limited to, violations that involve:

- Willfully causing a licensee to be in violation of NRC requirements.
- Willfully taking action that would have caused a licensee to be in violation of NRC requirements but the action did not do so because it was detected and corrective action was taken.
- Recognizing a violation of procedural requirements and willfully not taking corrective action.
- Willfully defeating alarms which have safety significance.
- Unauthorized abandoning of reactor controls.
- Dereliction of duty.
- Falsifying records required by NRC regulations or by the facility license.
- Willfully providing, or causing a licensee to provide, an NRC inspector or investigator with inaccurate or incomplete information on a matter material to the NRC.
- Willfully withholding safety significant information rather than making such information known to appropriate supervisory or technical personnel in the licensee's organization.
- Submitting false information and as a result gaining unescorted access to a nuclear power plant.
- Willfully providing false data to a licensee by a contractor or other person who provides test or other services, when the data affects the licensee's compliance with 10 CFR Part 50, Appendix B, or other regulatory requirement.
- Willfully providing false certification that components meet the requirements of their intended use, such as ASME Code.
- Willfully supplying, by contractors of equipment for transportation of radioactive material, casks that do not comply with their certificates of compliance.
- Willfully performing unauthorized bypassing of required reactor or other facility safety systems.
- Willfully taking actions that violate Technical Specification Limiting Conditions for Operation or other license conditions (enforcement action for a willful violation will not be taken if that violation is the result of action taken following the NRC's decision to forego enforcement of the Technical Specification or other license condition or if the operator meets the requirements of 10 CFR 50.54 (x), (i.e., unless the operator acted unreasonably considering all the relevant circumstances surrounding the emergency.)

Normally, some enforcement action is taken against a licensee for violations caused by significant acts of wrongdoing by its employees, contractors, or contractors' employees. In deciding whether to issue an enforcement action to an unlicensed person as well as to the licensee, the NRC recognizes that judgments will have to be made on a case by case basis. In making these decisions, the NRC will consider factors such as the following:

1. The level of the individual within the organization.

2. The individual's training and experience as well as knowledge of the potential consequences of the wrongdoing.

3. The safety consequences of the misconduct.

4. The benefit to the wrongdoer, e.g., personal or corporate gain.

5. The degree of supervision of the individual, i.e., how closely is the individual monitored or audited, and the likelihood of detection (such as a radiographer working independently in the field as contrasted with a team activity at a power plant).

6. The employer's response, e.g., disciplinary action taken.

7. The attitude of the wrongdoer, e.g., admission of wrongdoing, acceptance of responsibility.

8. The degree of management responsibility or culpability.

9. Who identified the misconduct.

Any proposed enforcement action involving individuals must be issued with the concurrence of the Deputy Executive Director. The particular sanction to be used should be determined on a case-by-case basis.[11] Notices of Violation and Orders are examples of enforcement actions that may be appropriate against individuals. The administrative action of a Letter of Reprimand may also be considered. In addition, the NRC may issue Demands for Information to gather information to enable it to determine whether an order or other enforcement action should be issued.

Orders to NRC-licensed reactor operators may involve suspension for a specified period, modification, or revocation of their individual licenses. Orders to unlicensed individuals might include provisions that would:

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[11]Except for individuals subject to civil penalties under section 206 of the Energy Reorganization Act of 1974, as amended, the NRC will not normally impose a civil penalty against an individual. However, section 234 of the Atomic Energy Act (AEA) gives the Commission authority to impose civil penalties on "any person." "Person" is broadly defined in Section 11s of the AEA to include individuals, a variety of organizations, and any representatives or agents. This gives the Commission authority to impose civil penalties on employees of licensees or on separate entities when a violation of a requirement directly imposed on them is committed.

- Prohibit involvement in NRC licensed activities for a specified period of time (normally the period of suspension would not exceed 5 years) or until certain conditions are satisfied, e.g., completing specified training or meeting certain qualifications.

- Require notification to the NRC before resuming work in licensed activities.

- Require the person to tell a prospective employer or customer engaged in licensed activities that the person has been subject to an NRC order.

In the case of a licensed operator's failure to meet applicable fitness-for-duty requirements (10 CFR 55.53(j)), the NRC may issue a Notice of Violation or a civil penalty to the Part 55 licensee, or an order to suspend, modify, or revoke the Part 55 license. These actions may be taken the first time a licensed operator fails a drug or alcohol test, that is, receives a confirmed positive test that exceeds the cutoff levels of 10 CFR Part 26 or the facility licensee's cutoff levels, if lower. However, normally only a Notice of Violation will be issued for the first confirmed positive test in the absence of aggravating circumstances such as errors in the performance of licensed duties or evidence of prolonged use. In addition, the NRC intends to issue an order to suspend the Part 55 license for up to 3 years the second time a licensed operator exceeds those cutoff levels. In the event there are less than 3 years remaining in the term of the individual's license, the NRC may consider not renewing the individual's license or not issuing a new license after the three year period is completed. The NRC intends to issue an order to revoke the Part 55 license the third time a licensed operator exceeds those cutoff levels. A licensed operator or applicant who refuses to participate in the drug and alcohol testing programs established by the facility licensee or who is involved in the sale, use, or possession of an illegal drug is also subject to license suspension, revocation, or denial.

In addition, the NRC may take enforcement action against a licensee that may impact an individual, where the conduct of the individual places in question the NRC's reasonable assurance that licensed activities will be properly conducted. The NRC may take enforcement action for reasons that would warrant refusal to issue a license on an original application. Accordingly, appropriate enforcement actions may be taken regarding matters that raise issues of integrity, competence, fitness-for-duty, or other matters that may not necessarily be a violation of specific Commission requirements.

In the case of an unlicensed person, whether a firm or an individual, an order modifying the facility license may be issued to require (1) the removal of the person from all licensed activities for a specified period of time or indefinitely, (2) prior notice to the NRC before using the person in licensed activities, or (3) the licensee to provide notice of the issuance of such an order to other persons involved in licensed activities making reference inquiries. In addition, orders to employers might require retraining, additional oversight, or independent verification of activities performed by the person, if the person is to be involved in licensed activities.

IX. INACCURATE AND INCOMPLETE INFORMATION

A violation of the regulations involving the submittal of incomplete and/or inaccurate information, whether or not considered a material false statement, can result in the full range of enforcement sanctions. The labeling of a communication failure as a material false statement will be made on a case-by-case basis and will be reserved for egregious violations. Violations involving inaccurate or incomplete information or the failure to provide significant information identified by a licensee normally will be categorized based on the guidance herein, in Section IV, "Significance of Violations," and in Supplement VII.

The Commission recognizes that oral information may in some situations be inherently less reliable than written submittals because of the absence of an opportunity for reflection and management review. However, the Commission must be able to rely on oral communications from licensee officials concerning significant information. Therefore, in determining whether to take enforcement action for an oral statement, consideration may be given to factors such as (1) the degree of knowledge that the communicator should have had, regarding the matter, in view of his or her position, training, and experience; (2) the opportunity and time available prior to the communication to assure the accuracy or completeness of the information; (3) the degree of intent or negligence, if any, involved; (4) the formality of the communication; (5) the reasonableness of NRC reliance on the information; (6) the importance of the information which was wrong or not provided; and (7) the reasonableness of the explanation for not providing complete and accurate information.

Absent at least careless disregard, an incomplete or inaccurate unsworn oral statement normally will not be subject to enforcement action unless it involves significant information provided by a licensee official. However, enforcement action may be taken for an unintentionally incomplete or inaccurate oral statement provided to the NRC by a licensee official or others on behalf of a licensee, if a record was made of the oral information and provided to the licensee thereby permitting an opportunity to correct the oral information, such as if a transcript of the communication or meeting summary containing the error was made available to the licensee and was not subsequently corrected in a timely manner.

When a licensee has corrected inaccurate or incomplete information, the decision to issue a Notice of Violation for the initial inaccurate or incomplete information normally will be dependent on the circumstances, including the ease of detection of the error, the timeliness of the correction, whether the NRC or the licensee identified the problem with the communication, and whether the NRC relied on the information prior to the correction. Generally, if the matter was promptly identified and corrected by the licensee prior to reliance by the NRC, or before the NRC raised a question about the information, no enforcement action will be taken for the initial inaccurate or incomplete information. On the other hand, if the misinformation is identified after the NRC relies on it, or after some question is raised regarding the accuracy of the information, then some enforcement action normally will be taken even if it is in fact corrected. However, if the initial submittal was accurate when made but later turns out to be erroneous because of newly discovered information or advance in technology, a citation normally would not be appropriate if, when the new information became available or the advancement in technology was made, the initial submittal was corrected.

The failure to correct inaccurate or incomplete information which the licensee does not identify as significant normally will not constitute a separate violation. However, the circumstances surrounding the failure to correct may be considered relevant to the determination of enforcement action for the initial inaccurate or incomplete statement. For example, an unintentionally inaccurate or incomplete submission may be treated as a more severe matter if the licensee later determines that the initial submittal was in error and does not correct it or if there were clear opportunities to identify the error. If information not corrected was recognized by a licensee as significant, a separate citation may be made for the failure to provide significant information. In any event, in serious cases where the licensee's actions in not correcting or providing information raise questions about its commitment to safety or its fundamental trustworthiness, the Commission may exercise its authority to issue orders modifying, suspending, or revoking the license. The Commission recognizes that enforcement determinations must be made on a case-by-case basis, taking into consideration the issues described in this section.

X. ENFORCEMENT ACTION AGAINST NON-LICENSEES

The Commission's enforcement policy is also applicable to non-licensees, including contractors and subcontractors, holders of NRC approvals, e.g., certificates of compliance, early site permits, standard design certificates, quality assurance program approvals, or applicants for any of them, and to employees of any of the foregoing, who knowingly provide components, equipment, or other goods or services that relate to a licensee's activities subject to NRC regulation. The prohibitions and sanctions for any of these persons who engage in deliberate misconduct or knowing submission of incomplete or inaccurate information are provided in the rule on deliberate misconduct, e.g., 10 CFR 30.10 and 50.5.

Contractors who supply products or services provided for use in nuclear activities are subject to certain requirements designed to ensure that the products or services supplied that could affect safety are of high quality. Through procurement contracts with licensees, suppliers may be required to have quality assurance programs that meet applicable requirements, e.g., 10 CFR Part 50, Appendix B, and 10 CFR Part 71, Subpart H. Contractors supplying certain products or services to licensees are subject to the requirements of 10 CFR Part 21 regarding reporting of defects in basic components.

When inspections determine that violations of NRC requirements have occurred, or that contractors have failed to fulfill contractual commitments (e.g., 10 CFR Part 50, Appendix B) that could adversely affect the quality of a safety significant product or service, enforcement action will be taken. Notices of Violation and civil penalties will be used, as appropriate, for licensee failures to ensure that their contractors have programs that meet applicable requirements.

Notices of Violation will be issued for contractors who violate 10 CFR Part 21. Civil penalties will be imposed against individual directors or responsible officers of a contractor organization who knowingly and consciously fail to provide the notice required by 10 CFR 21.21(d)(1). Notices of Violation or orders will be used against non-licensees who are subject to the specific requirements of Parts 71 and 72. Notices of Nonconformance will be used for contractors who fail to meet commitments related to NRC activities but are not in violation of specific requirements.

XI. REFERRALS TO THE DEPARTMENT OF JUSTICE

Alleged or suspected criminal violations of the Atomic Energy Act (and of other relevant Federal laws) are referred to the Department of Justice (DOJ) for investigation. Referral to the DOJ does not preclude the NRC from taking other enforcement action under this policy. However, enforcement actions will be coordinated with the DOJ in accordance with the Memorandum of Understanding between the NRC and the DOJ, (53 FR 50317; December 14, 1988).

XII. PUBLIC DISCLOSURE OF ENFORCEMENT ACTIONS

Enforcement actions and licensees' responses, in accordance with 10 CFR 2.790, are publicly available for inspection. In addition, press releases are generally issued for orders and civil penalties and are issued at the same time the order or proposed imposition of the civil penalty is issued. In addition, press releases are usually issued when a proposed civil penalty is withdrawn or substantially mitigated by some amount. Press releases are not normally issued for Notices of Violation that are not accompanied by orders or proposed civil penalties.

XIII. REOPENING CLOSED ENFORCEMENT ACTIONS

If significant new information is received or obtained by NRC which indicates that an enforcement sanction was incorrectly applied, consideration may be given, dependent on the circumstances, to reopening a closed enforcement action to increase or decrease the severity of a sanction or to correct the record. Reopening decisions will be made on a case-by-case basis, are expected to occur rarely, and require the specific approval of the Deputy Executive Director.

SUPPLEMENTS - VIOLATION EXAMPLES

This section provides examples of violations in each of four severity levels as guidance in determining the appropriate severity level for violations in each of eight activity areas (reactor operations, Part 50 facility construction, safeguards, health physics, transportation, fuel cycle and materials operations, miscellaneous matters, and emergency preparedness).

SUPPLEMENT I--REACTOR OPERATIONS

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of reactor operations.

A. ***Severity Level I*** **- Violations involving for example:**

1. A Safety Limit, as defined in 10 CFR 50.36 and the Technical Specifications being exceeded;
2. A system[12] designed to prevent or mitigate a serious safety event not being able to perform its intended safety function[13] when actually called upon to work;
3. An accidental criticality; or
4. A licensed operator at the controls of a nuclear reactor, or a senior operator directing licensed activities, involved in procedural errors which result in, or exacerbate the consequences of, an alert or higher level emergency and who, as a result of subsequent testing, receives a confirmed positive test result for drugs or alcohol.

B. ***Severity Level II*** **- Violations involving for example:**

1. A system designed to prevent or mitigate serious safety events not being able to perform its intended safety function;
2. A licensed operator involved in the use, sale, or possession of illegal drugs or the consumption of alcoholic beverages, within the protected area; or
3. A licensed operator at the control of a nuclear reactor, or a senior operator directing licensed activities, involved in procedural errors and who, as a result of subsequent testing, receives a confirmed positive test result for drugs or alcohol.

C. ***Severity Level III*** **- Violations involving for example:**

1. A significant failure to comply with the Action Statement for a Technical Specification Limiting Condition for Operation where the appropriate action was not taken within the required time, such as:

(a) In a pressurized water reactor, in the applicable modes, having one high-pressure safety injection pump inoperable for a period in excess of that allowed by the action statement; or

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[12]The term "system" as used in these supplements, includes administrative and managerial control systems, as well as physical systems.

[13]"Intended safety function" means the total safety function, and is not directed toward a loss of redundancy. A loss of one subsystem does not defeat the intended safety function as long as the other subsystem is operable.

(b) In a boiling water reactor, one primary containment isolation valve inoperable for a period in excess of that allowed by the action statement.

2. A system designed to prevent or mitigate a serious safety event not being able to perform its intended function under certain conditions (e.g., safety system not operable unless offsite power is available; materials or components not environmentally qualified).

3. Inattentiveness to duty on the part of licensed personnel;

4. Changes in reactor parameters that cause unanticipated reductions in margins of safety;

5. A non-willful compromise of an application, test, or examination required by 10 CFR Part 55 that:

(a) In the case of initial operator licensing, contributes to an individual being granted an operator or a senior operator license, or

(b) In the case of requalification, contributes to an individual being permitted to perform the functions of an operator or a senior operator.

6. A licensee failure to conduct adequate oversight of contractors resulting in the use of products or services that are of defective or indeterminate quality and that have safety significance;

7. A licensed operator's confirmed positive test for drugs or alcohol that does not result in a Severity Level I or II violation;

8. Equipment failures caused by inadequate or improper maintenance that substantially complicates recovery from a plant transient;

9. A failure to obtain prior Commission approval required by 10 CFR 50.59 for a change, in which the consequence of the change, is evaluated as having low to moderate, or greater safety significance (i.e., white, yellow, or red) by the SDP;

10. The failure to update the FSAR as required by 10 CFR 50.71(e) where the unupdated FSAR was used in performing a 10 CFR 50.59 evaluation for a change to the facility or procedures, implemented without prior Commission approval, that results in a condition evaluated as having low to moderate, or greater safety significance (i.e., white, yellow, or red) by the SDP; or

11. The failure to make a report required by 10 CFR 50.72 or 50.73 associated with any Severity Level III violation.

D. *Severity Level IV* - **Violations involving for example:**

1. A less significant failure to comply with the Action Statement for a Technical Specification Limiting Condition for Operation where the appropriate action was not taken within the required time, such as:

(a) In a pressurized water reactor, a 5 percent deficiency in the required volume of the condensate storage tank; or

(b) In a boiling water reactor, one subsystem of the two independent MSIV leakage control subsystems inoperable;

2. A non-willful compromise of an application, test, or examination required by 10 CFR Part 55 that:

(a) In the case of initial operator licensing, is discovered and reported to the NRC before an individual is granted an operator or a senior operator license, or

(b) In the case of requalification, is discovered and reported to the NRC before an individual is permitted to perform the functions of an operator or a senior operator, or

(c) Constitutes more than minor concern.

3. A failure to meet regulatory requirements that have more than minor safety or environmental significance;

4. A failure to make a required Licensee Event Report;

5. Violations of 10 CFR 50.59 that result in conditions evaluated as having very low safety significance (i.e., green) by the SDP; or

6. A failure to update the FSAR as required by 10 CFR 50.71(e) in cases where the erroneous information is not used to make an unacceptable change to the facility or procedures.

E. *Minor* - **Violations involving for example:**

A failure to meet 10 CFR 50.59 requirements where there was not a reasonable likelihood that the change requiring 10 CFR 50.59 evaluation would ever require Commission review and approval prior to implementation. In the case of a 10 CFR 50.71(e) violation, where a failure to update the FSAR would not have a material impact on safety or licensed activities.

SUPPLEMENT II--PART 50 FACILITY CONSTRUCTION

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of Part 50 facility construction.

A. *Severity Level I* - Violations involving structures or systems that are completed[14] in such a manner that they would not have satisfied their intended safety related purpose.

B. *Severity Level II* - **Violations involving for example:**

1. A breakdown in the Quality Assurance (QA) program as exemplified by deficiencies in construction QA related to more than one work activity (e.g., structural, piping, electrical, foundations). These deficiencies normally involve the licensee's failure to conduct adequate audits or to take prompt corrective action on the basis of such audits and normally involve multiple examples of deficient construction or construction of unknown quality due to inadequate program implementation; or

2. A structure or system that is completed in such a manner that it could have an adverse effect on the safety of operations.

C. *Severity Level III* - **Violations involving for example:**

1. A deficiency in a licensee QA program for construction related to a single work activity (e.g., structural, piping, electrical, or foundations). This significant deficiency normally involves the licensee's failure to conduct adequate audits or to take prompt corrective action on the basis of such audits, and normally involves multiple examples of deficient construction or construction of unknown quality due to inadequate program implementation;

2. A failure to confirm the design safety requirements of a structure or system as a result of inadequate preoperational test program implementation; or

3. A failure to make a required 10 CFR 50.55(e) report.

D. *Severity Level IV* - Violations involving failure to meet regulatory requirements including one or more Quality Assurance Criterion not amounting to Severity Level I, II, or III violations that have more than minor safety or environmental significance.

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[14]The term "completed" as used in this supplement means completion of construction including review and acceptance by the construction QA organization.

SUPPLEMENT III--SAFEGUARDS

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of safeguards.

A. ***Severity Level I* - Violations involving for example:**

1. An act of radiological sabotage in which the security system did not function as required and, as a result of the failure, there was a significant event, such as:

(a) A Safety Limit, as defined in 10 CFR 50.36 and the Technical Specifications, was exceeded;

(b) A system designed to prevent or mitigate a serious safety event was not able to perform its intended safety function when actually called upon to work; or

(c) An accidental criticality occurred;

2. The theft, loss, or diversion of a formula quantity[15] of special nuclear material (SNM); or

3. Actual unauthorized production of a formula quantity of SNM.

B. ***Severity Level II* - Violations involving for example:**

1. The entry of an unauthorized individual[16] who represents a threat into a vital area[17] from outside the protected area;

2. The theft, loss or diversion of SNM of moderate strategic significance[18] in which the security system did not function as required; or

3. Actual unauthorized production of SNM.

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[15]See 10 CFR 73.2 for the definition of "formula quantity."

[16]The term "unauthorized individual" as used in this supplement means someone who was not authorized for entrance into the area in question, or not authorized to enter in the manner entered.

[17]The phrase "vital area" as used in this supplement includes vital areas and material access areas.

[18]See 10 CFR 73.2 for the definition of "special nuclear material of moderate strategic significance."

C. *Severity Level III* - **Violations involving for example:**

1. A failure or inability to control access through established systems or procedures, such that an unauthorized individual (i.e., not authorized unescorted access to protected area) could easily gain undetected access[19] into a vital area from outside the protected area;

2. A failure to conduct any search at the access control point or conducting an inadequate search that resulted in the introduction to the protected area of firearms, explosives, or incendiary devices and reasonable facsimiles thereof that could significantly assist radiological sabotage or theft of strategic SNM;

3. A failure, degradation, or other deficiency of the protected area intrusion detection or alarm assessment systems such that an unauthorized individual who represents a threat could predictably circumvent the system or defeat a specific zone with a high degree of confidence without insider knowledge, or other significant degradation of overall system capability;

4. A significant failure of the safeguards systems designed or used to prevent or detect the theft, loss, or diversion of strategic SNM;

5. A failure to protect or control classified or safeguards information considered to be significant while the information is outside the protected area and accessible to those not authorized access to the protected area;

6. A significant failure to respond to an event either in sufficient time to provide protection to vital equipment or strategic SNM, or with an adequate response force; or

7. A failure to perform an appropriate evaluation or background investigation so that information relevant to the access determination was not obtained or considered and as a result a person, who would likely not have been granted access by the licensee, if the required investigation or evaluation had been performed, was granted access.

D. *Severity Level IV* - **Violations involving for example:**

1. A failure or inability to control access such that an unauthorized individual (i.e., authorized to protected area but not to vital area) could easily gain undetected access into a vital area from inside the protected area or into a controlled access area;

2. A failure to respond to a suspected event in either a timely manner or with an adequate response force;

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[19]In determining whether access can be easily gained, factors such as predictability, identifiability, and ease of passage should be considered.

3. A failure to implement 10 CFR Parts 25 and 95 with respect to the information addressed under Section 142 of the Act, and the NRC approved security plan relevant to those parts;

4. A failure to conduct a proper search at the access control point;

5. A failure to properly secure or protect classified or safeguards information inside the protected area that could assist an individual in an act of radiological sabotage or theft of strategic SNM where the information was not removed from the protected area;

6. A failure to control access such that an opportunity exists that could allow unauthorized and undetected access into the protected area but that was neither easily or likely to be exploitable;

7. A failure to conduct an adequate search at the exit from a material access area;

8. A theft or loss of SNM of low strategic significance that was not detected within the time period specified in the security plan, other relevant document, or regulation; or

9. Other violations that have more than minor safeguards significance.

SUPPLEMENT IV--HEALTH PHYSICS (10 CFR PART 20)

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of health physics, 10 CFR Part 20.[20]

A. *Severity Level I* - Violations involving for example:

1. A radiation exposure during any year of a worker in excess of 25 rems total effective dose equivalent, 75 rems to the lens of the eye, or 250 rads to the skin of the whole body, or to the feet, ankles, hands or forearms, or to any other organ or tissue;

2. A radiation exposure over the gestation period of the embryo/fetus of a declared pregnant woman in excess of 2.5 rems total effective dose equivalent;

3. A radiation exposure during any year of a minor in excess of 2.5 rems total effective dose equivalent, 7.5 rems to the lens of the eye, or 25 rems to the skin of the whole body, or to the feet, ankles, hands or forearms, or to any other organ or tissue;

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[20]Personnel overexposures and associated violations incurred during a life-saving or other emergency response effort will be treated on a case-by-case basis.

4. An annual exposure of a member of the public in excess of 1.0 rem total effective dose equivalent;

5. A release of radioactive material to an unrestricted area at concentrations in excess of 50 times the limits for members of the public as described in 10 CFR 20.1302(b)(2)(i); or

6. Disposal of licensed material in quantities or concentrations in excess of 10 times the limits of 10 CFR 20.2003.

B. *Severity Level II* - Violations involving for example:

1. A radiation exposure during any year of a worker in excess of 10 rems total effective dose equivalent, 30 rems to the lens of the eye, or 100 rems to the skin of the whole body, or to the feet, ankles, hands or forearms, or to any other organ or tissue;

2. A radiation exposure over the gestation period of the embryo/fetus of a declared pregnant woman in excess of 1.0 rem total effective dose equivalent;

3. A radiation exposure during any year of a minor in excess of 1 rem total effective dose equivalent; 3.0 rems to the lens of the eye, or 10 rems to the skin of the whole body, or to the feet, ankles, hands or forearms, or to any other organ or tissue;

4. An annual exposure of a member of the public in excess of 0.5 rem total effective dose equivalent;

5. A release of radioactive material to an unrestricted area at concentrations in excess of 10 times the limits for members of the public as described in 10 CFR 20.1302(b)(2)(i) (except when operation up to 0.5 rem a year has been approved by the Commission under §20.1301(c));

6. Disposal of licensed material in quantities or concentrations in excess of five times the limits of 10 CFR 20.2003; or

7. A failure to make an immediate notification as required by 10 CFR 20.2202 (a)(1) or (a)(2).

C. *Severity Level III* - Violations involving for example:

1. A radiation exposure during any year of a worker in excess of 5 rems total effective dose equivalent, 15 rems to the lens of the eye, or 50 rems to the skin of the whole body or to the feet, ankles, hands or forearms, or to any other organ or tissue;

2. A radiation exposure over the gestation period of the embryo/fetus of a declared pregnant woman in excess of 0.5 rem total effective dose equivalent (except when doses are in accordance with the provisions of §20.1208(d));

3. A radiation exposure during any year of a minor in excess of 0.5 rem total effective dose equivalent; 1.5 rems to the lens of the eye, or 5 rems to the skin of the whole body, or to the feet, ankles, hands or forearms, or to any other organ or tissue;

4. An annual exposure of a member of the public in excess of 0.1 rem total effective dose equivalent (except when operation up to 0.5 rem a year has been approved by the Commission under §20.1301(c));

5. A release of radioactive material to an unrestricted area at concentrations in excess of two times the effluent concentration limits referenced in 10 CFR 20.1302(b)(2)(i) (except when operation up to 0.5 rem a year has been approved by the Commission under Section 20.1301(c));

6. A failure to make a 24-hour notification required by 10 CFR 20.2202(b) or an immediate notification required by 10 CFR 20.2201(a)(1)(i);

7. A substantial potential for exposures or releases in excess of the applicable limits in 10 CFR 20.1001-20.2401 whether or not an exposure or release occurs;

8. Disposal of licensed material not covered in Severity Levels I or II;

9. A release for unrestricted use of contaminated or radioactive material or equipment that poses a realistic potential for exposure of the public to levels or doses exceeding the annual dose limits for members of the public;

10. Conduct of licensee activities by a technically unqualified person; or

11. A violation involving failure to secure, or maintain surveillance over, licensed material that:

(a) involves licensed material in any aggregate quantity greater than 1000 times the quantity specified in Appendix C to Part 20; or

(b) involves licensed material in any aggregate quantity greater than 10 times the quantity specified in Appendix C to Part 20, where such failure is accompanied by the absence of a functional program to detect and deter security violations that includes training, staff awareness, detection (including auditing), and corrective action (including disciplinary action); or

(c) results in a substantial potential for exposures or releases in excess of the applicable limits in Part 20.

D. ***Severity Level IV*** **- Violations involving for example:**

1. Exposures in excess of the limits of 10 CFR 20.1201, 20.1207, or 20.1208 not constituting Severity Level I, II, or III violations;

2. A release of radioactive material to an unrestricted area at concentrations in excess of the limits for members of the public as referenced in 10 CFR 20.1302(b)(2)(i) (except when operation up to 0.5 rem a year has been approved by the Commission under §20.1301(c));

3. A radiation dose rate in an unrestricted or controlled area in excess of 0.002 rem in any 1 hour (2 millirem/hour) or 50 millirems in a year;

4. Failure to maintain and implement radiation programs to keep radiation exposures as low as is reasonably achievable;

5. Doses to a member of the public in excess of any EPA generally applicable environmental radiation standards, such as 40 CFR Part 190;

6. A failure to make the 30-day notification required by 10 CFR 20.2201(a)(1)(ii) or 20.2203(a);

7. A failure to make a timely written report as required by 10 CFR 20.2201(b), 20.2204, or 20.2206;

8. A failure to report an exceedance of the dose constraint established in 10 CFR 20.1101(d) or a failure to take corrective action for an exceedance, as required by 10 CFR 20.1101(d);

9. Any other matter that has more than a minor safety, health, or environmental significance; or

10. A violation involving an isolated failure to secure, or maintain surveillance over, licensed material that is not otherwise characterized in Example IV.C.11 and that involves licensed material in any aggregate quantity greater than 10 times the quantity specified in Appendix C to Part 20, provided that: (i) the material is labeled as radioactive or located in an area posted as containing radioactive materials; and (ii) such failure occurs despite a functional program to detect and deter security violations that includes training, staff awareness, detection (including auditing), and corrective action (including disciplinary action).

E. *Minor* - **Violations involving for example:**

A violation involving an isolated failure to secure, or maintain surveillance over, licensed material in an aggregate quantity that does not exceed 10 times the quantity specified in Appendix C to Part 20.

SUPPLEMENT V--TRANSPORTATION

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of NRC transportation requirements.[21]

A. *Severity Level I* - **Violations involving for example:**

1. Failure to meet transportation requirements that resulted in loss of control of radioactive material with a breach in package integrity such that the material caused a radiation exposure to a member of the public and there was clear potential for the public to receive more than .1 rem to the whole body;

2. Surface contamination in excess of 50 times the NRC limit; or

3. External radiation levels in excess of 10 times the NRC limit.

B. *Severity Level II* - **Violations involving for example:**

1. Failure to meet transportation requirements that resulted in loss of control of radioactive material with a breach in package integrity such that there was a clear potential for the member of the public to receive more than .1 rem to the whole body;

2. Surface contamination in excess of 10, but not more than 50 times the NRC limit;

3. External radiation levels in excess of five, but not more than 10 times the NRC limit; or

4. A failure to make required initial notifications associated with Severity Level I or II violations.

C. *Severity Level III* - **Violations involving for example:**

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[21]Some transportation requirements are applied to more than one licensee involved in the same activity such as a shipper and a carrier. When a violation of such a requirement occurs, enforcement action will be directed against the responsible licensee which, under the circumstances of the case, may be one or more of the licensees involved.

1. Surface contamination in excess of five but not more than 10 times the NRC limit;

2. External radiation in excess of one but not more than five times the NRC limit;

3. Any noncompliance with labeling, placarding, shipping paper, packaging, loading, or other requirements that could reasonably result in the following:

(a) A significant failure to identify the type, quantity, or form of material;

(b) A failure of the carrier or recipient to exercise adequate controls; or

(c) A substantial potential for either personnel exposure or contamination above regulatory limits or improper transfer of material; or

4. A failure to make required initial notification associated with Severity Level III violations.

D. *Severity Level IV* - Violations involving for example:

1. A breach of package integrity without external radiation levels exceeding the NRC limit or without contamination levels exceeding five times the NRC limits;

2. Surface contamination in excess of but not more than five times the NRC limit;

3. A failure to register as an authorized user of an NRC-Certified Transport package;

4. A noncompliance with shipping papers, marking, labeling, placarding, packaging or loading not amounting to a Severity Level I, II, or III violation;

5. A failure to demonstrate that packages for special form radioactive material meets applicable regulatory requirements;

6. A failure to demonstrate that packages meet DOT Specifications for 7A Type A packages; or

7. Other violations that have more than minor safety or environmental significance.

SUPPLEMENT VI--FUEL CYCLE AND MATERIALS OPERATIONS

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of fuel cycle and materials operations.

A. *Severity Level I* - **Violations involving for example:**

1. Radiation levels, contamination levels, or releases that exceed 10 times the limits specified in the license;

2. A system designed to prevent or mitigate a serious safety event not being operable when actually required to perform its design function;

3. A nuclear criticality accident;

4. Failure to use a properly prepared written directive as required by 10 CFR 35.40; or failure to develop, implement, or maintain procedures for administrations requiring a written directive as required by 10 CFR 35.41; that results in a death or serious injury (e.g., substantial organ impairment);

5. A safety limit, as defined in 10 CFR 76.4, the Technical Safety Requirements, or the application being exceeded; or

6. Significant injury or loss of life due to a loss of control over licensed or certified activities, including chemical processes that are integral to the licensed or certified activity, whether radioactive material is released or not.

B. *Severity Level II* - **Violations involving for example:**

1. Radiation levels, contamination levels, or releases that exceed five times the limits specified in the license;

2. A system designed to prevent or mitigate a serious safety event being inoperable;

3. A substantial programmatic failure to implement written directives or procedures for administrations requiring a written directive, such as a failure of the licensee's procedures to address one or more of the elements in 10 CFR 35.40 or 35.41, or a failure to train personnel in those procedures, that results in a medical event;

4. A failure to establish, implement, or maintain all criticality controls (or control systems) for a single nuclear criticality scenario when a critical mass of fissile material was present or reasonably available, such that a nuclear criticality accident was possible; or

5. The potential for a significant injury or loss of life due to a loss of control over licensed or certified activities, including chemical processes that are integral to the licensed or certified activity, whether radioactive material is released or not (e.g., movement of liquid UF_6 cylinder by unapproved methods).

C. *Severity Level III* - **Violations involving for example:**

1. Possession or use of unauthorized equipment or materials in the conduct of licensee activities which degrades safety;

2. Use of radioactive material on humans where such use is not authorized;

3. Conduct of licensed activities by a technically unqualified or uncertified person;

4. A substantial potential for exposures, radiation levels, contamination levels, or releases, including releases of toxic material caused by a failure to comply with NRC regulations, from licensed or certified activities in excess of regulatory limits;

5. A substantial programmatic failure to implement written directives or procedures for administrations requiring a written directive, such as a failure of the licensee's procedures to address one or more of the elements in 10 CFR 35.40 or 35.41, or a failure to train personnel in those procedures, that does not result in a medical event. Failure to report a medical event. A programmatic weakness in the implementation of written directives or procedures for administrations requiring a written directive, whether or not a medical event occurs;

6. A failure, during radiographic operations, to have present at least two qualified individuals or to use radiographic equipment, radiation survey instruments, and/or personnel monitoring devices as required by 10 CFR Part 34;

7. A failure to submit an NRC Form 241 as required by 10 CFR 150.20;

8. A failure to receive required NRC approval prior to the implementation of a change in licensed activities that has radiological or programmatic significance, such as, a change in ownership; lack of an RSO or replacement of an RSO with an unqualified individual; a change in the location where licensed activities are being conducted, or where licensed material is being stored where the new facilities do not meet the safety guidelines; or a change in the quantity or type of radioactive material being processed or used that has radiological significance;

9. A significant failure to meet decommissioning requirements including a failure to notify the NRC as required by regulation or license condition, substantial failure to meet decommissioning standards, failure to conduct and/or complete decommissioning activities in accordance with regulation or license condition, or failure to meet required schedules without adequate justification;

10. A significant failure to comply with the action statement for a Technical Safety Requirement Limiting Condition for Operation where the appropriate action was not taken within the required time, such as:

(a) In an autoclave, where a containment isolation valve is inoperable for a period in excess of that allowed by the action statement; or

(b) Cranes or other lifting devices engaged in the movement of cylinders having inoperable safety components, such as redundant braking systems, or other safety devices for a period in excess of that allowed by the action statement;

11. A system designed to prevent or mitigate a serious safety event:

(a) Not being able to perform its intended function under certain conditions (e.g., safety system not operable unless utilities available, materials or components not according to specifications); or

(b) Being degraded to the extent that a detailed evaluation would be required to determine its operability;

12. Changes in parameters that cause unanticipated reductions in margins of safety;

13. A significant failure to meet the requirements of 10 CFR 76.68, including a failure such that a required certificate amendment was not sought;

14. A failure of the certificate holder to conduct adequate oversight of contractors resulting in the use of products or services that are of defective or indeterminate quality and that have safety significance;

15. Equipment failures caused by inadequate or improper maintenance that substantially complicates recovery from a plant transient;

16. A failure to establish, maintain, or implement all but one criticality control (or control systems) for a single nuclear criticality scenario when a critical mass of fissile material was present or reasonably available, such that a nuclear criticality accident was possible; or

17. A failure, during radiographic operations, to stop work after a pocket dosimeter is found to have gone off-scale, or after an electronic dosimeter reads greater than 200 mrem, and before a determination is made of the individual's actual radiation exposure.

D. ***Severity Level IV*** **- Violations involving for example:**

1. A failure to maintain patients hospitalized who have cobalt-60, cesium-137, or iridium-192 implants or to conduct required leakage or contamination tests, or to use properly calibrated equipment;

2. Other violations that have more than minor safety or environmental significance;

3. Failure to use a properly prepared written directive as required by 10 CFR 35.40; or failure to develop, implement, or maintain procedures for administrations requiring a written directive as required by 10 CFR 35.41, whether or not a medical event occurs, provided that the failures: (1) are isolated; (2) do not demonstrate programmatic weaknesses in implementation; and (3) have limited consequences if a medical event is involved;

4. A failure to keep the records required by 10 CFR 35.32 or 35.33;

5. A less significant failure to comply with the Action Statement for a Technical Safety Requirement Limiting Condition for Operation when the appropriate action was not taken within the required time;

6. A failure to meet the requirements of 10 CFR 76.68 that does not result in a Severity Level I, II, or III violation;

7. A failure to make a required written event report, as required by 10 CFR 76.120(d)(2); or

8. A failure to establish, implement, or maintain a criticality control (or control system) for a single nuclear criticality scenario when the amount of fissile material available was not, but could have been sufficient to result in a nuclear criticality.

SUPPLEMENT VII--MISCELLANEOUS MATTERS

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations involving miscellaneous matters.

A. *Severity Level I* - Violations involving for example:

1. Inaccurate or incomplete information[22] that is provided to the NRC (a) deliberately with the knowledge of a licensee official that the information is incomplete or inaccurate, or (b) if the information, had it been complete and accurate at the time provided, likely would have resulted in regulatory action such as an immediate order required by the public health and safety;

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[22]In applying the examples in this supplement regarding inaccurate or incomplete information and records, reference should also be made to the guidance in Section IX, "Inaccurate and Incomplete Information," and to the definition of "licensee official" contained in Section IV.C.

2. Incomplete or inaccurate information that the NRC requires be kept by a licensee that is (a) incomplete or inaccurate because of falsification by or with the knowledge of a licensee official, or (b) if the information, had it been complete and accurate when reviewed by the NRC, likely would have resulted in regulatory action such as an immediate order required by public health and safety considerations;

3. Information that the licensee has identified as having significant implications for public health and safety or the common defense and security ("significant information identified by a licensee") and is deliberately withheld from the Commission;

4. Action by senior corporate management in violation of 10 CFR 50.7 or similar regulations against an employee;

5. A knowing and intentional failure to provide the notice required by 10 CFR Part 21; or

6. A failure to substantially implement the required fitness-for-duty program.[23]

B. ***Severity Level II*** **- Violations involving for example:**

1. Inaccurate or incomplete information that is provided to the NRC (a) by a licensee official because of careless disregard for the completeness or accuracy of the information, or (b) if the information, had it been complete and accurate at the time provided, likely would have resulted in regulatory action such as a show cause order or a different regulatory position;

2. Incomplete or inaccurate information that the NRC requires be kept by a licensee which is (a) incomplete or inaccurate because of careless disregard for the accuracy of the information on the part of a licensee official, or (b) if the information, had it been complete and accurate when reviewed by the NRC, likely would have resulted in regulatory action such as a show cause order or a different regulatory position;

3. "Significant information identified by a licensee" and not provided to the Commission because of careless disregard on the part of a licensee official;

4. An action by plant management or mid-level management in violation of 10 CFR 50.7 or similar regulations against an employee;

5. A failure to provide the notice required by 10 CFR Part 21;

[23]The example for violations for fitness-for-duty relate to violations of 10 CFR Part 26.

6. A failure to remove an individual from unescorted access who has been involved in the sale, use, or possession of illegal drugs within the protected area or take action for on duty misuse of alcohol, prescription drugs, or over-the-counter drugs;

7. A failure to take reasonable action when observed behavior within the protected area or credible information concerning activities within the protected area indicates possible unfitness for duty based on drug or alcohol use;

8. A deliberate failure of the licensee's Employee Assistance Program (EAP) to notify licensee's management when EAP's staff is aware that an individual's condition may adversely affect safety related activities; or

9. The failure of licensee management to take effective action in correcting a hostile work environment.

C. *Severity Level III* - Violations involving for example:

1. Incomplete or inaccurate information that is provided to the NRC (a) because of inadequate actions on the part of licensee officials but not amounting to a Severity Level I or II violation, or (b) if the information, had it been complete and accurate at the time provided, likely would have resulted in a reconsideration of a regulatory position or substantial further inquiry such as an additional inspection or a formal request for information;

2. Incomplete or inaccurate information that the NRC requires be kept by a licensee that is (a) incomplete or inaccurate because of inadequate actions on the part of licensee officials but not amounting to a Severity Level I or II violation, or (b) if the information, had it been complete and accurate when reviewed by the NRC, likely would have resulted in a reconsideration of a regulatory position or substantial further inquiry such as an additional inspection or a formal request for information;

3. Inaccurate or incomplete performance indicator (PI) data submitted to the NRC by a Part 50 licensee that would have caused a PI to change from green to either yellow or red; white to either yellow or red; or yellow to red.

4. A failure to provide "significant information identified by a licensee" to the Commission and not amounting to a Severity Level I or II violation;

5. An action by first-line supervision or other low-level management in violation of 10 CFR 50.7 or similar regulations against an employee;

6. An inadequate review or failure to review such that, if an appropriate review had been made as required, a 10 CFR Part 21 report would have been made;

7. A failure to complete a suitable inquiry on the basis of 10 CFR Part 26, keep records concerning the denial of access, or respond to inquiries concerning denials of access so that, as a result of the failure, a person previously denied access for fitness-for-duty reasons was improperly granted access;

8. A failure to take the required action for a person confirmed to have been tested positive for illegal drug use or take action for onsite alcohol use; not amounting to a Severity Level II violation;

9. A failure to assure, as required, that contractors have an effective fitness-for-duty program; or

10. Threats of discrimination or restrictive agreements which are violations under NRC regulations such as 10 CFR 50.7(f).

D. *Severity Level IV* - Violations involving for example:

1. Incomplete or inaccurate information that is provided to the NRC but not amounting to a Severity Level I, II, or III violation;

2. Information that the NRC requires be kept by a licensee and that is incomplete or inaccurate and of more than minor significance but not amounting to a Severity Level I, II, or III violation;

3. Inaccurate or incomplete performance indicator (PI) data submitted to the NRC by a Part 50 licensee that would have caused a PI to change from green to white.

4. An inadequate review or failure to review under 10 CFR Part 21 or other procedural violations associated with 10 CFR Part 21 with more than minor safety significance;

5. Violations of the requirements of Part 26 of more than minor significance;

6. A failure to report acts of licensed operators or supervisors pursuant to 10 CFR 26.73; or

7. Discrimination cases which, in themselves, do not warrant a Severity Level III categorization.

E. *Minor* - Violations involving for example:

Inaccurate or incomplete performance indicator (PI) data submitted to the NRC by a Part 50 licensee that would not have caused a PI to change color.

SUPPLEMENT VIII--EMERGENCY PREPAREDNESS

This supplement provides examples of violations in each of the four severity levels as guidance in determining the appropriate severity level for violations in the area of emergency preparedness. It should be noted that citations are not normally made for violations involving emergency preparedness occurring during emergency exercises. However, where exercises reveal (i) training, procedural, or repetitive failures for which corrective actions have not been taken, (ii) an overall concern regarding the licensee's ability to implement its plan in a manner that adequately protects public health and safety, or (iii) poor self critiques of the licensee's exercises, enforcement action may be appropriate.

A. *Severity Level I* - Violations involving for example:

In a general emergency, licensee failure to promptly (1) correctly classify the event, (2) make required notifications to responsible Federal, State, and local agencies, or (3) respond to the event (e.g., assess actual or potential offsite consequences, activate emergency response facilities, and augment shift staff)..

B. *Severity Level II* - Violations involving for example:

1. In a site emergency, licensee failure to promptly (1) correctly classify the event, (2) make required notifications to responsible Federal, State, and local agencies, or (3) respond to the event (e.g., assess actual or potential offsite consequences, activate emergency response facilities, and augment shift staff); or

2. A licensee failure to meet or implement more than one emergency planning standard involving assessment or notification.

C. *Severity Level III* - Violations involving for example:

1. In an alert, licensee failure to promptly (1) correctly classify the event, (2) make required notifications to responsible Federal, State, and local agencies, or (3) respond to the event (e.g., assess actual or potential offsite consequences, activate emergency response facilities, and augment shift staff); or

2. A licensee failure to meet or implement one emergency planning standard involving assessment or notification.

D. *Severity Level IV* - Violations involving for example:

A licensee failure to meet or implement any emergency planning standard or requirement not directly related to assessment and notification.

INTERIM ENFORCEMENT POLICIES

Interim Enforcement Policy for Generally Licensed Devices Containing Byproduct Material (10 CFR 31.5)

This section sets forth the interim enforcement policy that the NRC will follow to exercise enforcement discretion for certain violations of requirements in 10 CFR Part 31 for generally licensed devices containing byproduct material. It addresses violations that persons licensed pursuant to 10 CFR 31.5 identify and correct now, as well as during the initial cycle of the notice and response program contemplated by the proposed new requirements published in the Federal Register on December 2, 1998 (63 FR 66492), entitled "Requirements for Those Who Possess Certain Industrial Devices Containing Byproduct Material to Provide Requested Information".

Exercise of Enforcement Discretion

Under this interim enforcement policy, enforcement action normally will not be taken for violations of 10 CFR 31.5 if they are identified by the general licensee, and reported to the NRC if reporting is required, if the general licensee takes appropriate corrective action to address the specific violations and prevent recurrence of similar problems.

Exceptions

Enforcement action may be taken where there is: (a) failure to take appropriate corrective action to prevent recurrence of similar violations; (b) failure to respond and provide the information required by the notice and response program (if it becomes a final rule); (c) failure to provide complete and accurate information to the NRC; or (d) a willful violation, such as willfully disposing of generally licensed material in an unauthorized manner. Enforcement sanctions in these cases may include civil penalties as well as Orders to modify or revoke the authority to possess radioactive sources under the general license.

Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fitness-for-Duty Issues (10 CFR Part 26)

This section sets forth the interim enforcement policy that the NRC will follow to exercise enforcement discretion for certain violations of requirements in 10 CFR Part 26, Fitness-for-Duty Programs that occur after December 30, 2002. The NRC will also exercise enforcement discretion and normally not pursue past violations for insufficient suitable inquiries (where licensees failed to contact employers when individuals had worked for employers for less than 30 days) and past violations for failures to perform pre-access drug tests (where individuals were subject to a FFD program within the last 30 days) that occurred prior to December 30,2002. The policy, subject to subsequent Commission-approved associated policy, guidance, or regulation, is in effect until a final revision of 10 CFR Part 26 is issued and becomes effective.

Suitable Inquiry

The regulation in 10 CFR 26.3 requires that before granting an individual unescorted access, a licensee must conduct a suitable inquiry consisting of a "best-effort verification of employment history for the past five years, but in no case less than three years, obtained through contacts with previous employers to determine if a person was, in the past, tested positive for illegal drugs, subject to a plan for treating substance abuse, removed from, or made ineligible for activities within the scope of 10 CFR Part 26, or denied unescorted access at any other nuclear power plant or other employment in accordance with a fitness-for-duty policy."

The requirement does not provide an exception when an individual is reinstated at a licensee facility or transferred within a licensee corporation or to another licensee where there is little or no interruption in authorization. The term, "authorization," refers to a period during which an individual maintained unescorted access or was assigned to perform activities within the scope of Part 26. However, enforcement action will not normally be taken for failure to contact interim employers, if the following practice is adopted:

If the individual applicant's authorization has been interrupted for 30 calendar days or less and the individual's last authorization was terminated favorably, before granting authorization for unescorted access to the protected area of a nuclear power plant or assigning the individual to perform activities within the scope of Part 26, the licensee shall obtain and verify that a self-disclosure (i.e., a report of any drug- or alcohol-related arrests) for the period since the last authorization contains no potentially disqualifying FFD information, unless the individual was subject to a licensee-approved behavioral observation and arrest-reporting program throughout the period of interruption. Potentially disqualifying FFD information means information demonstrating that an individual has, during the period authorization was interrupted:

(1) Violated an employer's drug and alcohol testing policy;
(2) Used, sold, or possessed illegal drugs;

(3) Abused legal drugs;
(4) Subverted or attempted to subvert a drug or alcohol testing program;
(5) Refused to take a drug or alcohol test;
(6) Been subjected to a plan for substance abuse treatment (except for self-referral); or
(7) Had legal or employment action taken for alcohol or drug use.

The licensee shall also ensure that the individual has met FFD refresher training requirements.

The requirements also do not provide an exception for each licensee to conduct a suitable inquiry into an individual applicant's past five years of employment when an individual is reinstated at a licensee facility or transferred to another licensee facility. However, enforcement action will not normally be taken for failure to contact employers from the past five years, if the following practice is adopted:

Licensees may rely upon the information gathered by previous licensees regarding an individual applicant's past five years of employment to meet the suitable inquiry requirement.

The NRC may take enforcement action when a licensee does not follow these practices.

Pre-access Testing

The regulation in 10 CFR 26.24(a)(1) requires that a person be tested for drugs and alcohol "within 60 days prior to the initial granting of unescorted access to protected areas."

The requirement does not provide an exception when an individual is reinstated at a licensee facility or transferred within a licensee corporation or to another licensee where there is little or no interruption in authorization. However, enforcement action will not normally be taken for failure to conduct a pre-access test for alcohol and drugs, if the following practice is adopted:

If the individual applicant's authorization has been interrupted for 30 calendar days or less and the individual's last authorization was terminated favorably, in order to grant authorization for unescorted access to the protected area of a nuclear power plant or assigning the individual to perform activities within the scope of Part 26, the licensee shall:

(1) Obtain and verify that a self-disclosure for the past 30 days reveals no potentially disqualifying information, unless the individual was subject to a licensee-approved behavioral observation and arrest-reporting program throughout the period of interruption; and
(2) Ensure that the individual has met FFD refresher training requirements.

If the individual applicant's authorization has been interrupted for 31 days to 60 days and the individual's last authorization was terminated favorably, in order to grant authorization for unescorted access to the protected area of a nuclear power plant or assigning the individual to perform activities within the scope of Part 26, the licensee shall:

(1) Obtain and verify that a self-disclosure for the period since the last authorization contains no potentially disqualifying FFD information, unless the individual was subject to a licensee-approved behavioral observation and arrest-reporting program throughout the period of interruption;

(2) Within 5 working days of granting authorization, complete a suitable inquiry for the period since last authorization was terminated, unless the individual was subject to a licensee-approved behavioral observation and arrest-reporting program throughout the period of interruption;

(3) Verify that results of an alcohol test are negative and collect a specimen for drug testing, unless either a drug and alcohol test meeting the standards of Part 26 was performed within the past 60 days and results were negative or the individual was subject to a licensee-approved Part 26 FFD program that included random drug and alcohol testing throughout the period of interruption; and

(4) Ensure that the individual has met FFD refresher training requirements.

The NRC may take enforcement action when a licensee does not follow these practices.

Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)

This section sets forth the interim enforcement policy that the NRC will follow to exercise enforcement discretion for certain violations of requirements in 10 CFR 50.48, Fire protection (or fire protection license conditions) that are identified as a result of the transition to a new risk-informed, performance-based fire protection approach included in paragraph (c) of 10 CFR 50.48 and for certain existing identified noncompliances that reasonably may be resolved by compliance with 10 CFR 50.48(c). Paragraph (c) allows reactor licensees to voluntarily comply with the risk-informed, performance-based fire protection approaches in National Fire Protection Association (NFPA) Standard 805 (NFPA 805), "Performance-Based Standard For Fire Protection For Light Water Reactor Electric Generating Plants," 2001 Edition (with limited exceptions stated in the rule language).

For those noncompliances identified during the licensee's transition process, this enforcement discretion policy will be in effect for up to two years from the date of a licensee's letter of intent to adopt the requirements in 10 CFR 50.48(c) and will continue to be in place until NRC approval of the license amendment request to transition to 10 CFR 50.48(c). This discretion policy may be extended upon a request from the licensee with adequate justification.

If, after submitting the letter of intent to comply with 10 CFR 50.48(c) and before submitting the license amendment request, the licensee determines not to complete the transition to 10 CFR 50.48(c), the licensee must submit a letter stating their intent to retain their existing license basis and withdrawing their letter of intent to comply with 50.48(c). Any violations identified prior to the date of the above withdrawal letter will be eligible for discretion, provided they are resolved under the existing licensing basis and meet the criteria included in this policy for these violations. Violations identified after the date of the above withdrawal letter will be dispositioned in accordance with normal enforcement practices.

A. Noncompliances Identified During the Licensee's Transition Process

Under this interim enforcement policy, enforcement action normally will not be taken for a violation of 10 CFR 50.48(b) (or the requirements in a fire protection license condition) involving a problem such as in engineering, design, implementing procedures, or installation, if the violation is documented in an inspection report and it meets all of the following criteria:

(1) It was licensee-identified as a result of its voluntary initiative to adopt the risk-informed, performance-based fire protection program included under 10 CFR 50.48(c) or, if the NRC identifies the violation, it was likely in the NRC staff's view that the licensee would have identified the violation in light of the defined scope, thoroughness, and schedule of the licensee's transition to 10 CFR 50.48(c) provided the schedule reasonably provides for completion of the transition within two years of the date of the licensee's letter of intent to implement 10 CFR 50.48(c) or other period granted by NRC;

(2) It was corrected or will be corrected as a result of completing the transition to 10 CFR 50.48(c). Also, immediate corrective action and/or compensatory measures are taken within a reasonable time commensurate with the risk significance of the issue following identification (this action should involve expanding the initiative, as necessary, to identify other issues caused by similar root causes);

(3) It was not likely to have been previously identified by routine licensee efforts such as normal surveillance or quality assurance (QA) activities; and

(4) It was not willful.

The NRC may take enforcement action when these conditions are not met or when a violation that is associated with a finding of high safety significance is identified.

While the NRC may exercise discretion for violations meeting the required criteria where the licensee failed to make a required report to the NRC, a separate enforcement action will normally be issued for the licensee's failure to make a required report.

B. Existing Identified Noncompliances

In addition, licensees may have existing identified noncompliances that could reasonably be corrected under 10 CFR 50.48(c). For these noncompliances, the NRC is providing enforcement discretion for the implementation of corrective actions until the licensee has transitioned to 10 CFR 50.48(c) provided that the noncompliances meet all of the following criteria:

(1) The licensee has entered the noncompliance into their corrective action program and implemented appropriate compensatory measures,

(2) The noncompliance is not associated with a finding that the Reactor Oversight Process Significance Determination Process would evaluate as Red, or it would not be categorized at Severity Level I,

(3) The licensee submits a letter of intent by December 31, 2005, stating its intent to transition to 10 CFR 50.48(c).

After December 31, 2005, as addressed in (3) above, this enforcement discretion for implementation of corrective actions for existing identified noncompliances will not be available and the requirements of 10 CFR 50.48(b) (and any other requirements in fire protection license conditions) will be enforced in accordance with normal enforcement practices.

Interim Enforcement Policy Regarding the Use of Alternative Dispute Resolution

I. Introduction

A. Background

This section sets forth the interim enforcement policy that the NRC will follow to undertake a pilot program testing the use of Alternative Dispute Resolution (ADR) in the enforcement program.

B. Scope

The pilot program scope consists of the trial use of ADR for cases involving: (1) alleged discrimination for engaging in protected activity prior to an NRC investigation; and (2) both discrimination and other wrongdoing cases after the Office of Investigations has competed an investigation. Specific points in the enforcement process where ADR may be requested are specified below. Mediation will be the form of ADR typically utilized. Certain cases may only require facilitation, a process where the neutral's function is primarily to support the communication process rather than focusing on the parties reaching a settlement.

Note: Although the NRC's ADR program may cause the parties to negotiate issues which may also form the basis for a claim under Section 211 of the Energy Reorganization Act of 1974, as amended, the Department of Labor's (DOL) timeliness requirements for filing a claim are in no way altered by the NRC's program.

In cases involving an allegation of discrimination, any underlying technical issue will be treated as a separate issue, or concern, within the allegation program. The allegation program will be used to resolve concerns (typically safety concerns) and issues other than the discrimination complaint.

II. General

A. Responsibilities and Program Administration

The Director, OE, is responsible for the overall program. In addition, the Director, OE, will serve as the lead NRC negotiator for cases involving discrimination after OI completes an investigation. The Director, OE, may also designate the Deputy Director, OE, to act as the lead negotiator.

Regional Administrators are designated as the lead NRC negotiator for cases involving wrongdoing other than discrimination. The Regional Administrator may designate the Deputy Regional Administrator to act as the lead negotiator or the Director or Deputy Director, OE, may also serve as the lead negotiator for other wrongdoing cases.

The Program Administrator will provide program oversight and support for each region and headquarters program offices. Program and neutral evaluations will be provided to the Program Administrator. The Program Administrator may serve as the intake neutral for post investigation ADR. An ``intake neutral" develops information and processes information for mediation. As an intake neutral, the confidentiality provisions discussed below will apply.

The Office Allegation Coordinators (OACs) are normally a complainant's first substantive contact when a concern regarding discrimination is raised. As such, the OACs may serve as an intake neutral who develops information and processes the necessary information for mediation under Early ADR. The OAC has the option to refer the whistleblower to the third party neutral to process the necessary information for mediation under Early ADR. The confidentiality provisions in Section II.B.7 will apply to the OAC, third party intake neutral, and Program Administrator. The OAC will also process documentation necessary to operate the program.

B. General Rules/Principles

Unless specifically addressed in a subsequent section, the rules described in this section apply generally throughout the ADR program, regardless of where in the overall enforcement process the ADR sessions occur.

1. *Voluntary*. Use of the NRC ADR program is voluntary, and any participant may end the mediation at any time. The goal is to obtain an agreement satisfactory to all participants on issues in controversy.

2. *Neutral qualification*. Generally, a neutral should be knowledgeable and experienced with nuclear matters or labor and employment law. However, any neutral that is satisfactory to the parties is acceptable.

3. *Roster of neutrals*. OE will maintain a list of organizations from which services of neutrals could be obtained. The parties may select a mediator from any of these organizations; however, the parties are not required to use the organizations provided and any neutral mutually agreeable to the parties is acceptable.

4. *Mediator selection.* If the parties have not selected a mediator within fourteen days, the Program Administrator or OAC may propose a mediator for the parties' consideration.

5. *Neutrality.* Mediators are neutral. The role of the mediator is to provide an environment where all participants will have an opportunity to resolve their differences. The parties should each consult an attorney or other professional if any question of law, content of a proposed agreement on issues in controversy, or other issues exists.

   For Early ADR, the OAC or third party neutral will serve as an intake neutral. Should any party seek to discuss the NRC's enforcement ADR process in detail, the party should be referred to the OAC or third party neutral. The OAC will initiate discussion of the option to mediate and process the necessary documentation. Subsequently, for post investigation ADR, the program administrator or third party neutral will serve as the intake neutral. Due to the nature of conversations that typically occur between an intake neutral and the parties, these conversations will also be considered confidential.

6. *Mediation sessions.* Once selected by the parties and contracted by the OAC or third party intake neutral, the mediator will promptly contact each of the parties to discuss the mediation process under the Program, reconfirm party interest in proceeding, establish a date and location for the mediation session and obtain any other information s/he believes likely to be useful. The mediator will preside over all mediation sessions, and will be expected to complete the mediation within 90 days after referral unless the parties, and the NRC if not a party, agree otherwise. At the conclusion of the mediation, parties will be asked to fill out and submit an evaluation form for the mediator that will be sent to the Program Administrator.

   Normally, a settlement is expected to be reached and signed within 90 days from when the parties agree to attempt ADR. A principal reason for Early ADR is the quick resolution of the claim, thereby improving the safety conscious work environment (SCWE). If the parties cannot agree to a settlement within 90 days, the NRC must assume a settlement will not be reached and continue with the investigation and enforcement process. Where good cause is shown and all parties agree, the NRC may allow a small extension to the 90 day limit to allow for completion of a settlement agreement.

   Settlement agreements in Early ADR will not be final until 3 days after the agreement has been signed. Either party may reconsider the settlement agreement during the 3 day period. Subsequent concerns regarding implementation of the settlement agreement should be directed to the neutral, or if necessary, the OAC.

7. *Confidentiality.* The mediator will specifically inform all parties and other attendees that all mediation activities under the Program are subject to the confidentiality provisions of the Administrative Dispute Resolution Act, 5 U.S.C. 574; the Federal ADR Council's guidance document entitled "Confidentiality in Federal ADR Programs;" and the explicit confidentiality terms set forth in the Agreement to Begin Voluntary Mediation signed by the parties. The mediator will explain these confidentiality terms and offer to answer questions regarding them.

8. *Good Faith.* All participants will participate in good faith in the mediation process and explore potentially feasible options that could lead to the management or resolution of issues in controversy.

9. *Not legal representation.* A mediator is not a legal representative or legal counsel. The mediator will not represent any party in the instant case or any future proceeding or matter relating to the issues in controversy in this case. The mediator is not either party's lawyer and no party should rely on the mediator for legal advice.

10. *Mediator Fees.* If Early ADR (defined below) is utilized, the NRC, subject to the availability of funds, will pay the mediator's entire fee. For cases where a licensee requests ADR subsequent to the completion of an OI report, the licensee requesting ADR will pay half of the mediator's fee and the NRC, subject to the availability of funds, will pay half. The NRC will recover the mediator fees it pays through annual fees assessed to licensees under 10 CFR Part 171.

11. *Exceptions.* The only exception to the offering of Early ADR by the NRC will be abuse of the program, e.g., a large number of repetitive requests for ADR by a particular facility, contractor, or whistleblower. Should the NRC believe the ADR program has been abused in some manner by one of the parties potentially involved, the Director, OE will be notified.

    To maximize the potential use of the ADR pilot program, for cases after an OI investigation is completed, the NRC will at least consider negotiating a settlement with a licensee for any wrongdoing case if requested. However, there may be certain circumstances where it may not be appropriate for the NRC to engage in ADR.

12. *Number of settlement attempts.* Each case will be afforded a maximum of two attempts to reach a settlement on the same underlying issue through the use of ADR. An ``attempt" is defined as one or more mediated sessions conducted at a specific point in the NRC's enforcement process (generally within a 90 day period). However, in general, settlement at any time without the use of a neutral is not precluded by the ADR program.

13. *Finality.* Cases that reach a settlement (and are acceptable to the NRC), either in Early ADR or after an OI investigation is complete, constitute a final enforcement decision on the case by the NRC.

III. ADR Opportunities

A. Licensee Sponsored Programs

Licensees are encouraged to develop ADR programs of their own for use in conjunction with an employee concerns type program. If an employee who alleges retaliation for engaging in protected activity utilizes a licensee's program to settle the discrimination concern, either before or after contacting the NRC, the licensee may voluntarily report the settlement to the NRC as a settlement within the NRC's jurisdiction. If notified of the settlement prior to initiation of an investigation, the NRC will review the settlement for restrictive agreements potentially in violation of 10 CFR 50.7(f), or other, similar regulations. Assuming no such restrictive agreements exist, the NRC will not investigate or take enforcement action.

B. Early ADR

The term "Early ADR" refers to the use of ADR prior to an OI investigation. The parties to Early ADR will normally be the complainant and the licensee. If the complainant is an employee of a licensee contractor, the parties will be the complainant and the contractor. Generally, the Early ADR process will parallel and work in conjunction with the NRC allegation program.

The allegation process will be used through the determination of a prima facie case. If an Allegation Review Board (ARB) determines a prima facie case exists, the ARB will normally recommend the parties be offered the opportunity to use Early ADR. Exceptions to such a recommendation should be rare and be based solely on an identified and articulated abuse of the ADR process by a party who would be involved in the case under consideration. Exceptions will be approved by the Director, OE, prior to initiating an investigation based on denial of ADR.

Early ADR cases will be tracked in the Allegation Management System (AMS). However, the allegation process timeliness measurement will be stayed once the ARB determines that ADR should be offered until the point in time ADR is declined by either party or the case is settled.

When an agreement is reached, the mediator will record the terms of that agreement. The parties may sign the agreement at the mediation session, or any party may review the agreement with his/her attorney before the document is placed in final form and signed. However, as noted above, settlement agreements in Early ADR will not be final until at least 3 days after the agreement has been signed. No participant will hold the NRC liable for the results of the mediation, whether or not a resolution is reached.

A settlement agreement between the parties will be reviewed by the NRC. OE will coordinate the review with the Office of the General Counsel (OGC). The review will ensure that no restrictive agreements in violation of 10 CFR 50.7(f) or other NRC regulations are contained in the settlement and will normally be completed within 5 working days of receipt. Given an acceptable settlement, the NRC will not investigate or take enforcement action.

The NRC expects that parties to Early ADR will agree to some form of confidentiality. However, that agreement cannot extend to the reporting of any safety concerns potentially discussed during the ADR sessions if one of the parties desires to report the concern. Either party may report safety concerns discussed during ADR sessions to the NRC without regard to confidentiality agreements. Safety concerns and their disposition may be discussed between the parties if desired. In cases where an Early ADR negotiation is between a licensee contractor and the contractor's employee, the NRC expects the contractor to ensure the licensee is aware of any safety issues discussed during the negotiations.

In addition to the settlement agreement, the licensee should provide the NRC with any planned or completed actions relevant to the safety conscious work environment that the licensee has determined to be appropriate.

Generally no press release or other public announcement will be made by the NRC for cases settled by early ADR. However, all documents, including the proposed settlement agreement, submitted to the NRC will be official agency records, and while not generally publicly available, still subject to the Freedom of Information Act (FOIA).

Documents associated with processing an Early ADR case will not generally be publicly available, consistent with the allegation program. However, documents may be subject to the FOIA and may be released, subject to redaction, pursuant to a FOIA request.

Some negotiations may fail to settle the case. When a settlement is not reached, the appropriate intake neutral will be notified, typically by the mediator, and an ARB will determine the appropriate action in accordance with the allegation program.

C. Post-Investigation ADR

Post-investigation ADR refers to the use of ADR anytime after an OI investigation is complete and an enforcement panel concludes that pursuit of an enforcement action appears warranted. Generally, post- investigation ADR processes will parallel and work in conjunction with the NRC enforcement program.

After an investigation is complete, there are generally three issues that can be resolved using ADR; whether a violation occurred, the appropriate enforcement action, and the appropriate corrective actions for the violation(s). If the parties agree, any or all three may be considered in an ADR session.

Two different types of enforcement cases will be eligible for ADR after an investigation is complete, discrimination and other wrongdoing cases. ADR will normally be considered at three places in the enforcement process after OI has completed an investigation: (1) After an enforcement panel has concluded there is the need to continue pursuing potential enforcement action based on an OI case and prior to the conduct of a predecisional enforcement conference (PEC); (2) after the initial enforcement action is taken, typically a Notice of Violation (NOV) and potentially a proposed civil penalty; and (3) after imposition of a civil penalty and prior to a hearing request.

The parties to an ADR session after an OI investigation is complete will be the licensee and the NRC. Fees associated with the neutral will typically be divided between the NRC and the licensee, with each paying half of the total cost.

Settlement discussions are expected to be complete within 90 days of initiating ADR prior to a PEC. The NRC may withdraw from settlement discussions if negotiations have not been completed in a timely manner.

The terms of a settlement agreement will normally be confirmed by order. Typically, the specific terms of settlement will be agreed to during the negotiation. The staff will then incorporate appropriate terms into a confirmatory order, a draft of which will then be agreed to by the licensee prior to issuance.

If an attempt to resolve a case using ADR prior to the conduct of a PEC fails, a predecisional enforcement conference will normally be offered to the licensee. The PEC will be conducted as described in the Enforcement Policy.

For cases within the scope of the pilot program, after a panel concludes that a case warrants continuation of the enforcement process, the responsible region or office will contact the licensee and offer either a PEC or ADR. Consistent with the Enforcement Policy, a written response could be offered at the staff's discretion.

Public notification of the settlement will normally be a press release and the confirmatory order will be published in the *Federal Register.*

Confidentiality with the NRC as a party will be determined by the parties as allowed by the ADR Act.

1. Discrimination Cases

   Consistent with centralization of the discrimination enforcement process, the Director, Office of Enforcement, will normally negotiate for the NRC.

   Normally the NRC will coordinate participation of the complainant. While the complainant will not be a party to the ADR process after OI issues an investigation report, the NRC will typically seek the complainant's input to the process. Normally, the NRC will at least seek input from the complainant regarding suggested corrective actions aimed at improving the safety conscious work environment.

   OI reports (not including exhibits) will normally be provided to the licensee when the choice of ADR or a PEC is offered.

   A licensee may request ADR for discrimination violations based solely on a finding by DOL. However, the staff will not negotiate the finding by DOL. The appropriate enforcement sanction and corrective actions will be the typical focus of settlement discussions.

2. Other Than Discrimination Wrongdoing

   The regional administrator will normally be the principal negotiator for the NRC in ADR sessions on other wrongdoing cases. After imposition of a civil penalty or other order, the Director, Office of Enforcement and applicable regional administrator may determine that the Director would be the appropriate negotiator.

Typically, an enforcement panel will be conducted to discuss the NRC's specific interests in the case prior to the regional administrator attending the settlement discussions. A limited review of the settlement terms may be conducted in conjunction with the preparation of the confirmatory order.

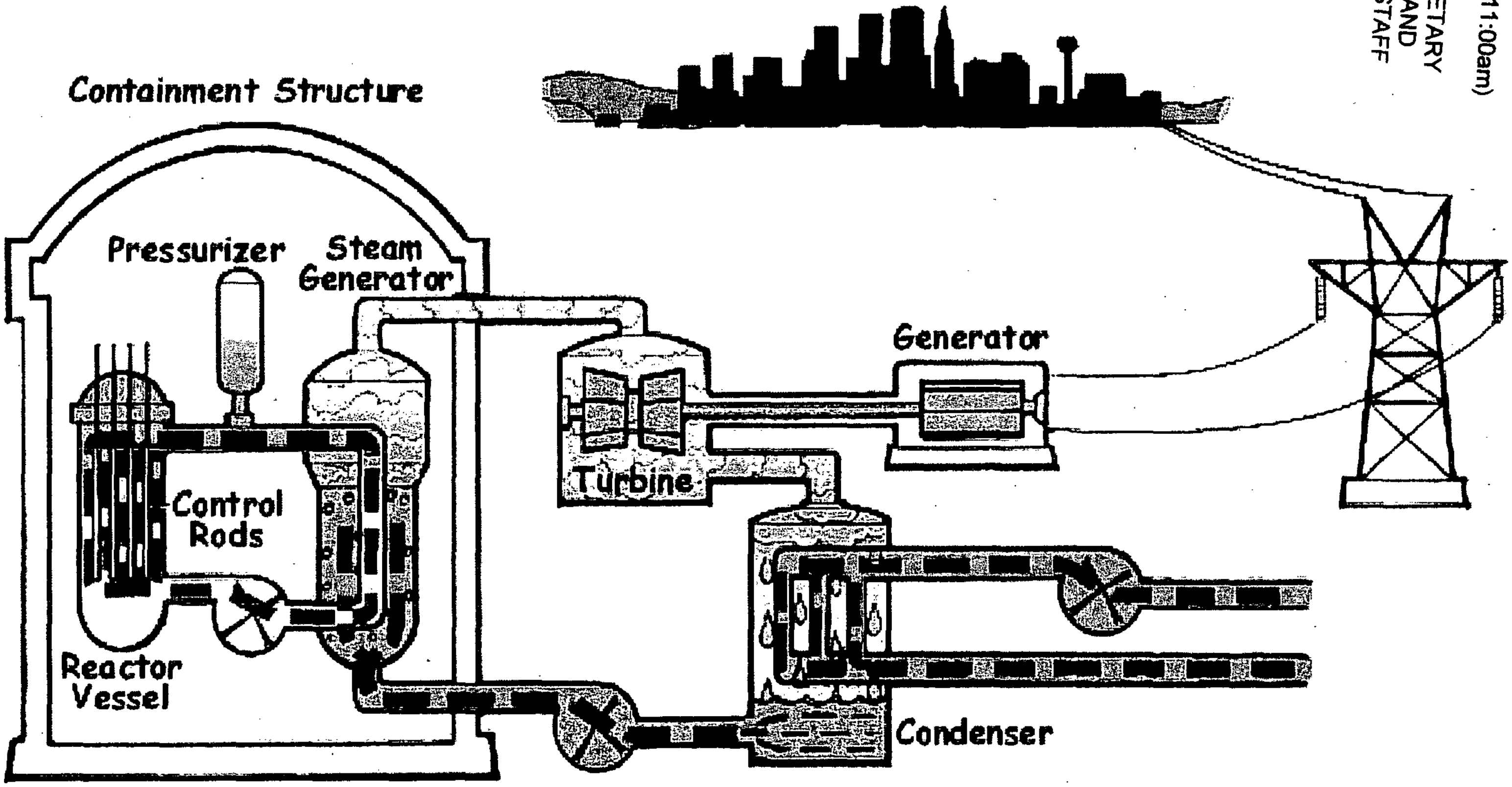
The OI report will not routinely be offered to the licensee prior to ADR. However, the OI report may be provided, as necessary, during the negotiations with the licensee.

IV. Integration With Traditional Enforcement Policy

A. Potential Future Enforcement Actions Civil Penalty Assessments

Section VI.C.2 of the Enforcement Policy provides the method for determination of a civil penalty amount. One aspect of the determination uses enforcement history as a factor. If the staff considers a civil penalty for a future escalated enforcement action, settlements under the enforcement ADR program occurring after a formal enforcement action is taken (e.g. an NOV is issued) may count as an enforcement case for purposes of determining whether identification credit is considered. Settlements occurring prior to an OI investigation will not count as previous enforcement. The status of settlement agreements occurring after an investigation is completed but prior to an NOV being issued will be established as part of the negotiation between the parties.

Pressurized Nuclear Reactor



RAS C-143

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In re DAVID GEISEN Staff Exhibit # 2
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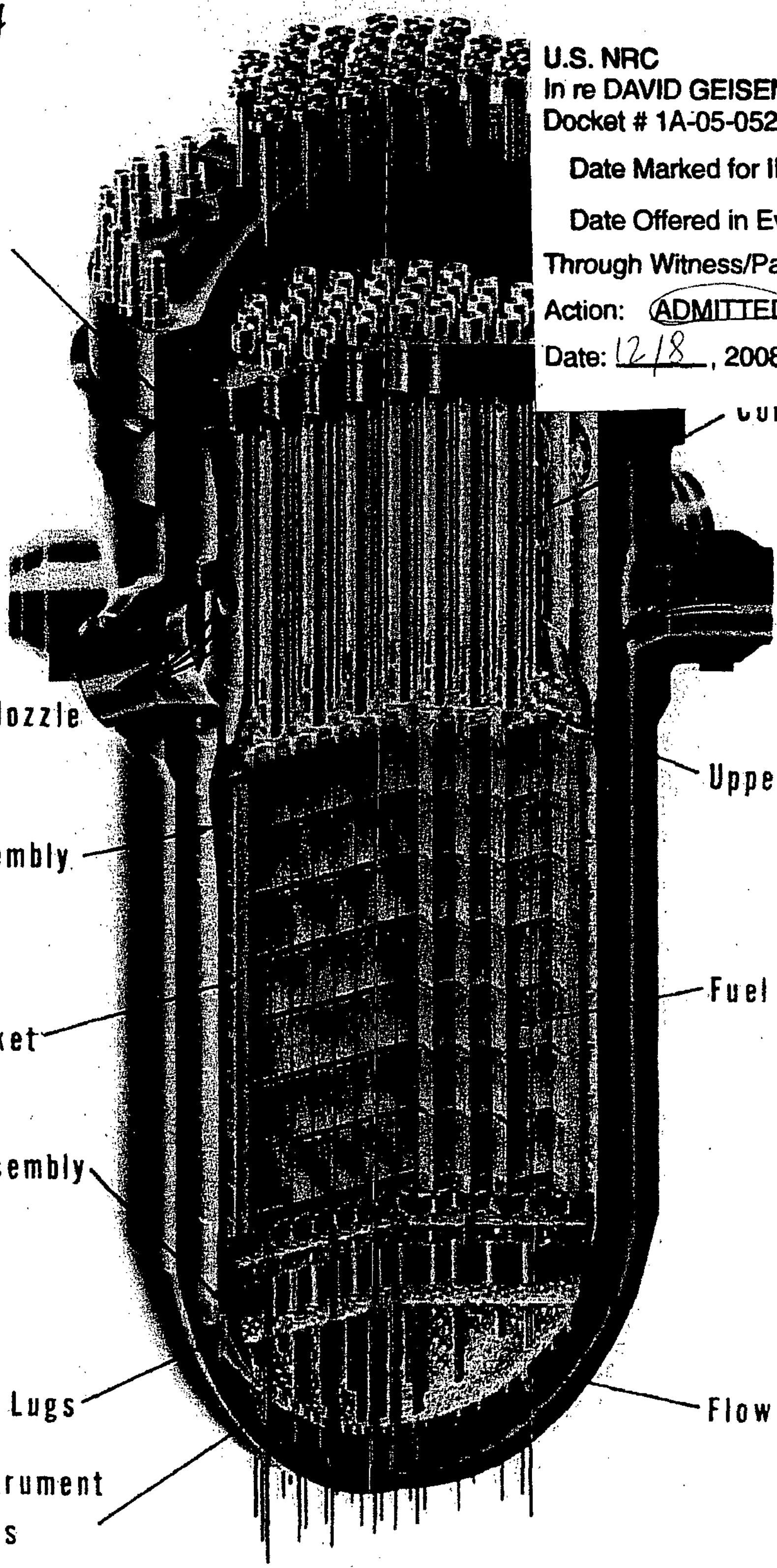

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PRESSURIZED WATER REACTOR
BABCOCK & WILCOX

DS02

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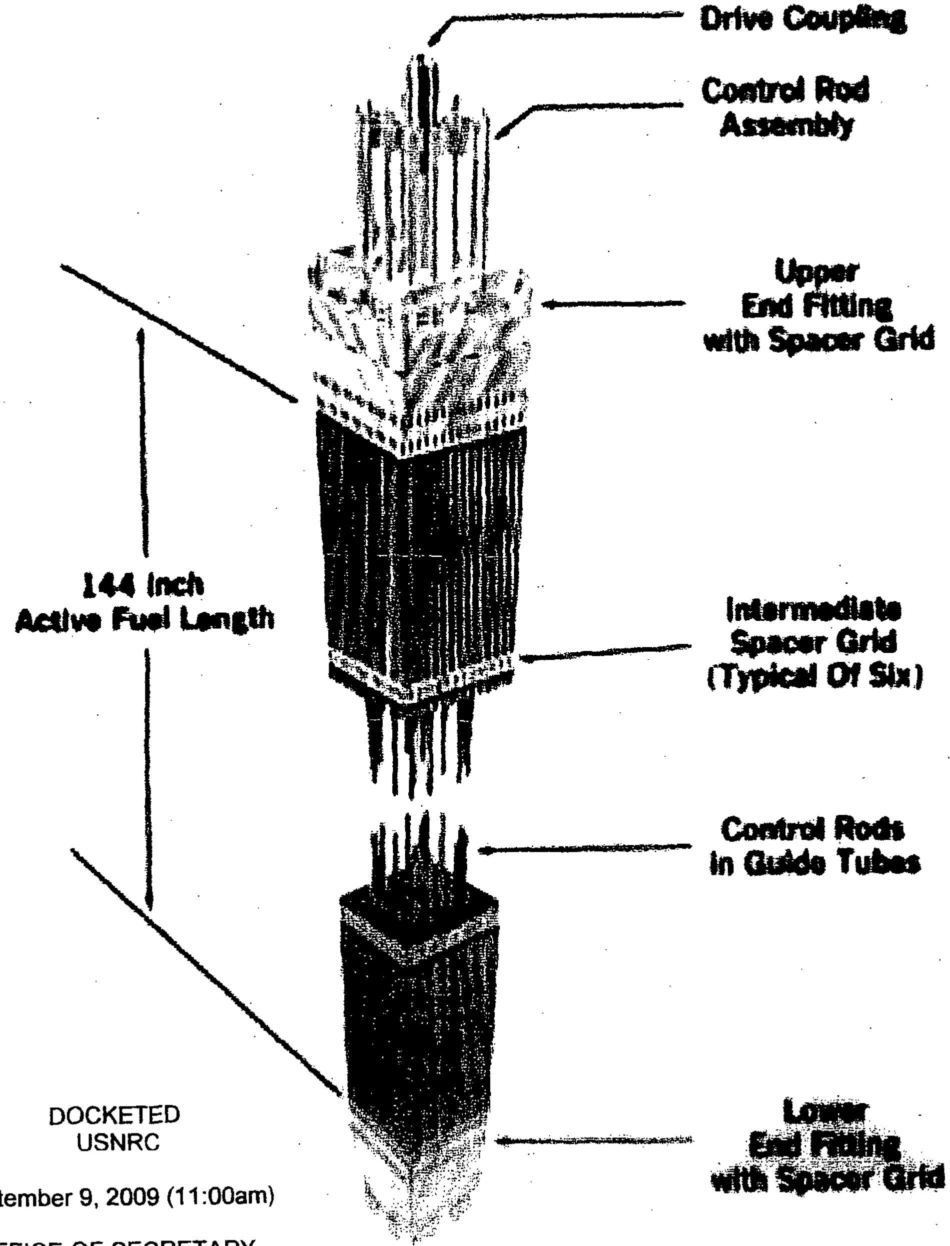
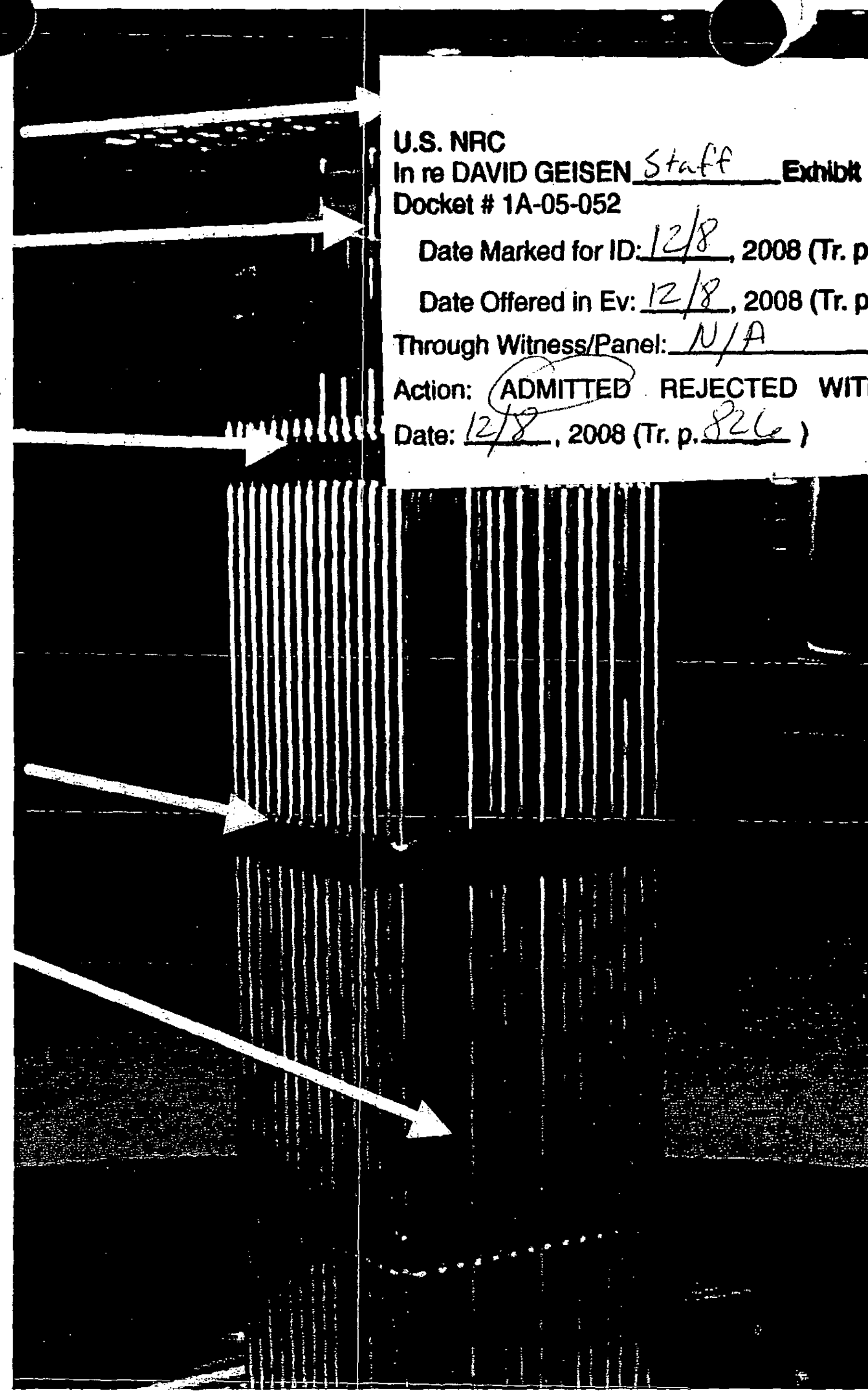
Figure 1-4. Fuel Assembly - Cutaway View





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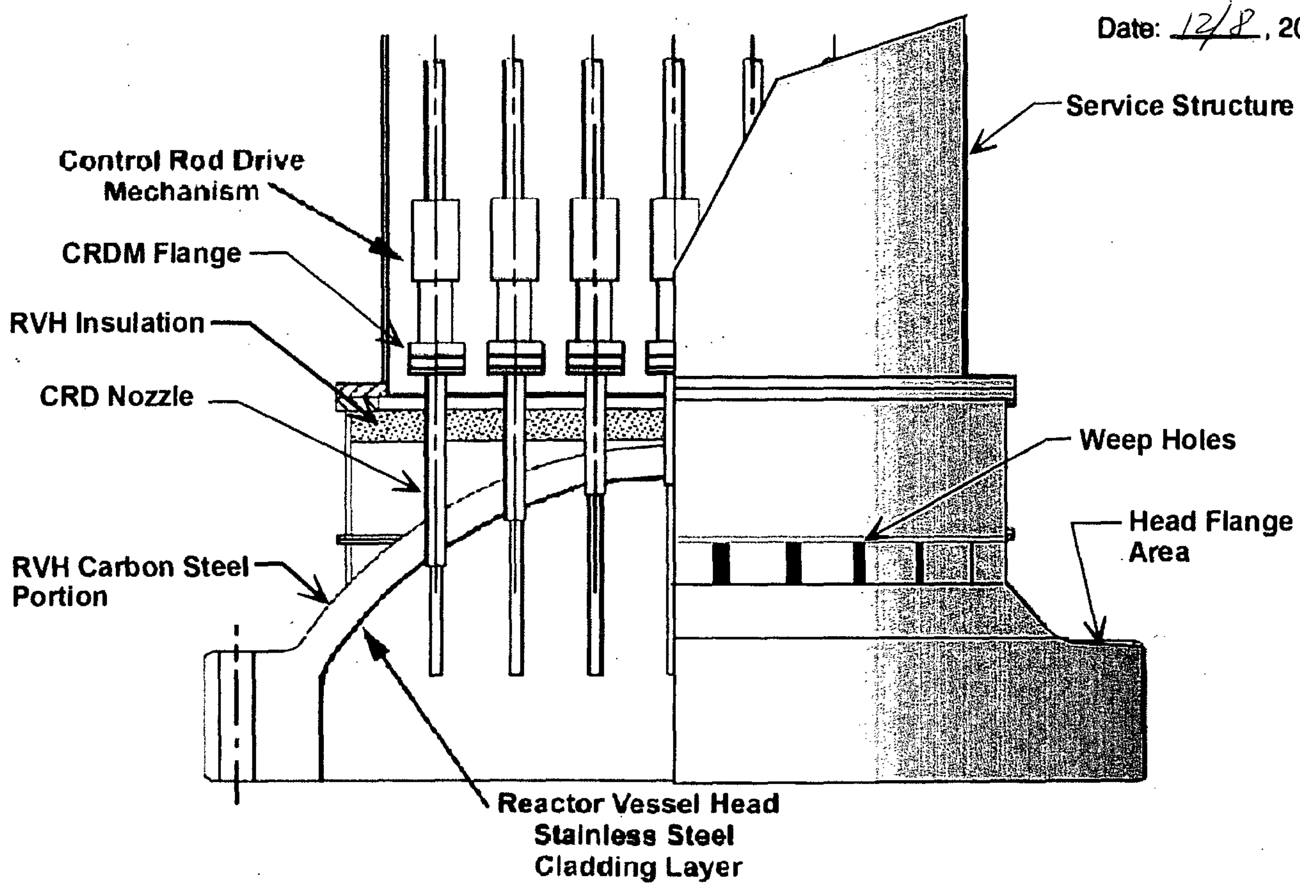
Reactor Vessel Head
Cross-Sectional View



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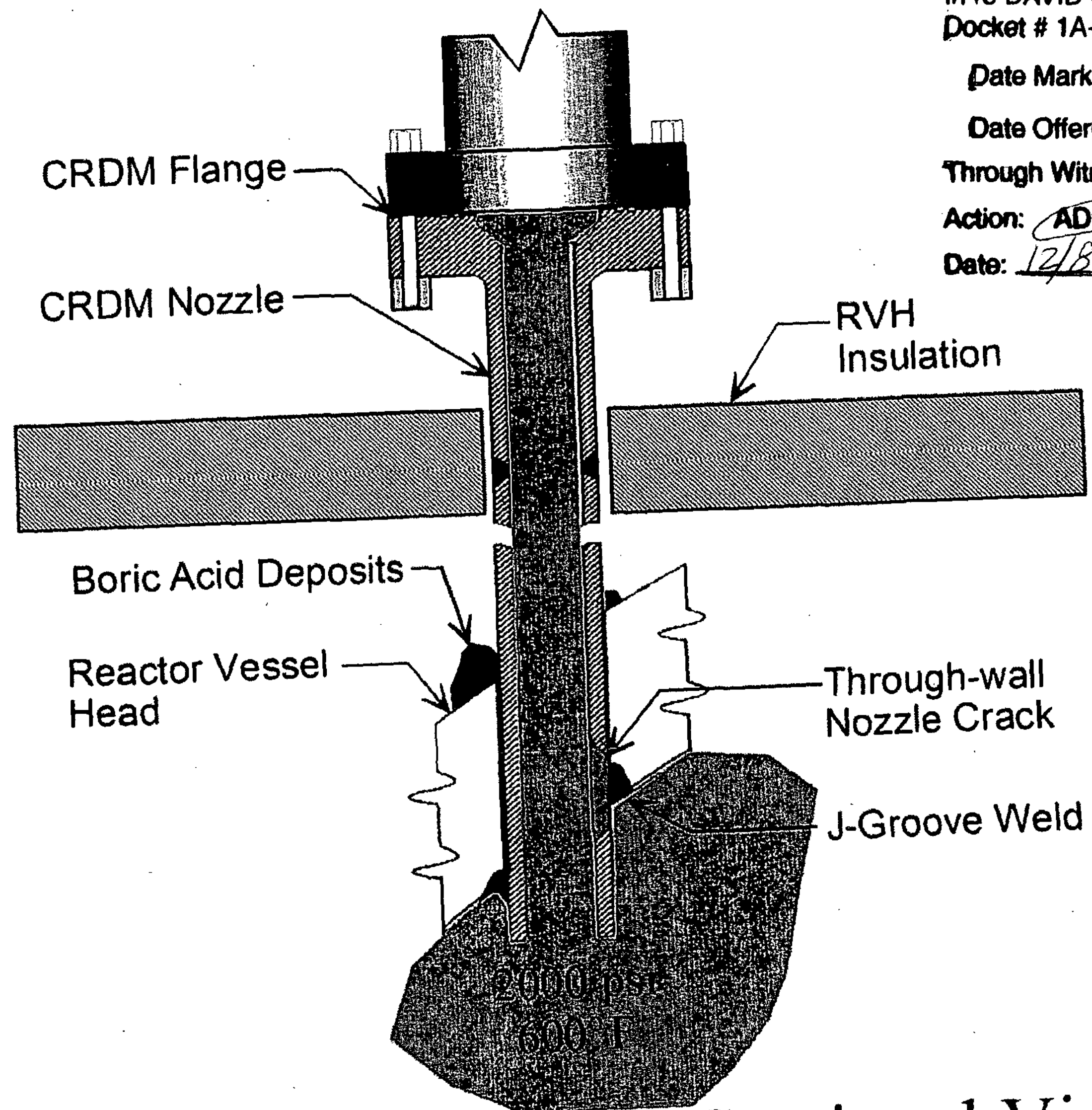

CRDM Nozzle Cross-Sectional View

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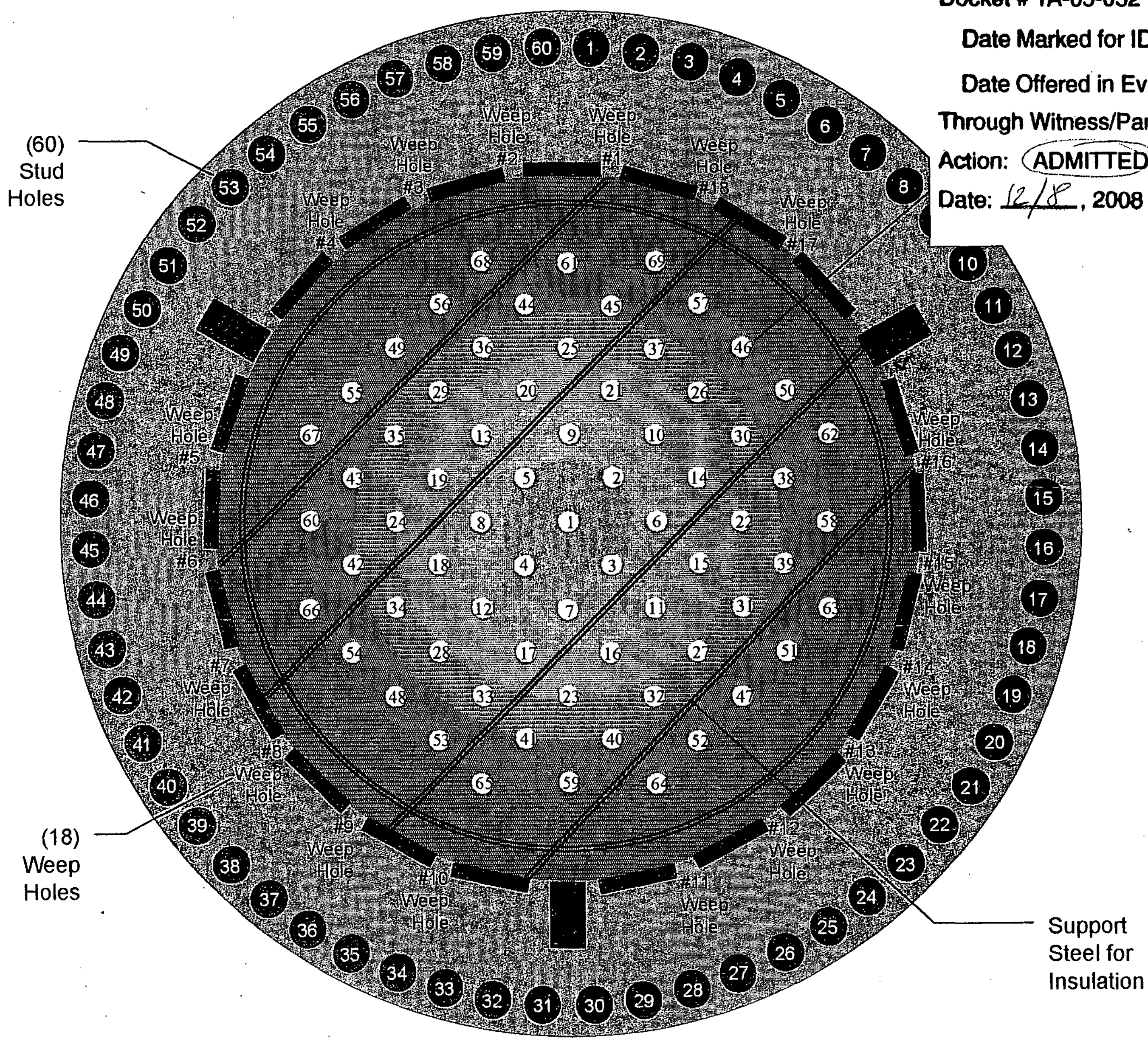
RAS C-148



Reactor Vessel Head Map

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OMB Control No.: 3150-0012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 3, 2001

NRC BULLETIN 2001-01: CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

Addressees

All holders of operating licenses for pressurized water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to:

(1) request that addressees provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements, and

(2) require that all addressees provide to the NRC a written response in accordance with the provisions of 10 CFR 50.54(f).

Background

The recent discoveries of cracked and leaking Alloy 600 VHP nozzles, including control rod drive mechanism (CRDM) and thermocouple nozzles, at four pressurized water reactors (PWRs) have raised concerns about the structural integrity of VHP nozzles throughout the PWR industry. Nozzle cracking at Oconee Nuclear Station Unit 1 (ONS1) in November 2000 and Arkansas Nuclear One Unit 1 (ANO1) in February 2001 was limited to axial cracking, an occurrence deemed to be of limited safety concern in the NRC staff's generic safety evaluation on the cracking of VHP nozzles, dated November 19, 1993. However, the discovery of circumferential cracking at Oconee Nuclear Station Unit 3 (ONS3) in February 2001 and Oconee Nuclear Station Unit 2 (ONS2) in April 2001 S particularly the large circumferential cracking identified in two CRDM nozzles at ONS3 S has raised concerns about the potential safety implications and prevalence of cracking in VHP nozzles in PWRs.

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NRC007-2015

As described in NRC Information Notice (IN) 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," dated April 30, 2001, Duke Energy Corporation (the licensee) performed a visual examination (VT-2) on the outer surface of the reactor pressure vessel (RPV) head at ONS3 to inspect for indications of borated water leakage, as part of normal surveillance during a planned maintenance outage. This visual examination followed cleaning of the RPV head during the prior outage to remove all existing boric acid deposits (from other sources such as leaking CRDM flanges) that could mask the identification of subsequent deposits that would be indicative of new or ongoing leakage. The VT-2 examination revealed small amounts of boric acid deposits (less than 1 cubic inch) at locations where the CRDM nozzles exit the RPV head for 9 of the 69 CRDM nozzles. Subsequent nondestructive examination (NDE) identified 47 recordable crack indications in the 9 degraded CRDM nozzles. The licensee initially characterized these flaws as being axial and a part of the RPV pressure boundary, or below-the-weld circumferential indications (which are not part of the RPV pressure boundary), and initiated repairs of the degraded areas.

Subsequent dye-penetrant testing (PT) of the repaired areas revealed the presence of additional indications in two of the nine degraded nozzles. While repairing the indications in these two nozzles, the licensee found that each nozzle had a circumferential crack that extended about 165° around the nozzle, above the weld (i.e., at a location that is part of the RPV pressure boundary). Further investigation and metallurgical examination identified that these cracks had initiated from the outside diameter (OD) of the CRDM nozzles. The circumferential crack in one of the nozzles was through-wall, and the crack in the other nozzle had pin hole indications on the nozzle inside diameter (ID). These cracks followed the contour of the weld profile.

The licensee stated that pre-repair ultrasonic testing (UT) examinations had identified indications in these areas, but that these indications had been misinterpreted as inconsequential craze cracking with unusual characteristics. The characterizations of these two nozzle indications were subsequently revised following the initial post-repair PT examinations. The licensee concluded that the root cause of the CRDM nozzle cracking was primary water stress corrosion cracking (PWSCC). The cracking initiated at the OD of the nozzles after cracking of the J-groove weld (see below) or adjacent heat-affected zone metal permitted coolant leakage into the annular region between the CRDM nozzle and the RPV head. This conclusion was based on metallurgical examinations, crack location and orientation, and finite element analyses.

The CRDM nozzles at ONS3 are approximately 5 feet long and are J-groove welded to the inner radius of the RPV head, with the lower end of each nozzle extending about 6 inches below the inside of the RPV head (see Attachment). The nozzles are constructed from 4-inch OD Alloy 600 Inconel procured in accordance with the requirements of Specification SB-167 to the 1965 Edition, including Addenda through the Summer 1967 Addenda, of Section II of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The weld preparation for the installation of each nozzle in the RPV head was accomplished by machining and buttering the J-groove with Alloy 182 weld metal. The RPV head was subsequently stress relieved and then the final machining of the CRDM penetrations, including the counterbore, was accomplished. Each nozzle was then machined to final dimensions to assure the appropriate design interference fit between the RPV head bore and the OD of the nozzle. The interference fit of the CRDM nozzles was made using a shrink fit process to install the CRDM nozzles. In this process, the nozzles were cooled to at least -140°F; they were then inserted into the closure head penetration, and the entire assembly was allowed to warm to room temperature (70°F minimum). The CRDM nozzles were tack welded and then permanently welded to the closure head using Alloy 182 weld metal. The manual shielded metal arc welding (SMAW) process was used for both the tack weld and the J-groove weld. During weld buildup, the weld was ground and PT inspected at each 9/32 inch of the weld. The final weld surface was ground and PT inspected.

The design and fabrication process for the VHPs in all PWR plants is similar to that described for ONS3.

Since the issuance of NRC IN 2001-05, circumferential cracking was identified in another CRDM nozzle, at ONS2. During a visual examination of the RPV head, Duke Energy Corporation identified boric acid deposits in the vicinity of four CRDM nozzles at ONS2. Subsequent UT examination identified a single CRDM nozzle with one OD-initiated circumferential crack, having a crack depth of 0.070 inch (~11% through-wall) and a length of 1.26 inches (~10% of the circumference).

Cracking due to PWSCC in PWR CRDM nozzles and other VHP nozzles fabricated from Alloy 600 is not a new issue; axial cracking in the CRDM nozzles has been identified since the late 1980s. In addition, numerous small-bore Alloy 600 nozzles and pressurizer heater sleeves have experienced leaks attributable to PWSCC. Generally, these components are exposed to high temperatures (greater than 550°F) and a primary water environment. However, circumferential cracking from the nozzle OD to the ID, above the weld, and cracking of the J-groove weld have not been previously identified in PWRs.

As described in Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997, an action plan was implemented by the NRC staff in 1991 to address PWSCC of Alloy 600 VHP nozzles at all operating U.S. PWRs. After reviewing safety assessments submitted by the industry and examining overseas inspection findings, the NRC staff concluded in its generic safety evaluation that CRDM nozzle and weld cracking in PWRs was not an immediate safety concern. The basis for this conclusion was that if PWSCC occurred (1) the cracks would be predominately axial in orientation, (2) the axial cracks would result in detectable leakage before catastrophic failure (with the expectation that CRDM nozzle cracking would result in a substantial volume of leaking coolant) and (3) the expected large amount of leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RPV head occurred. The safety evaluation identified concerns about potential circumferential cracking (which would need to be addressed on a plant-specific

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basis) as a consequence of high residual stresses resulting from initial manufacture and the impact of tube straightening that may have been needed after welding. The safety evaluation also noted the need for enhanced leakage monitoring.

The generic responses of licensees to GL 97-01 were predicated on the development of susceptibility ranking models to relate the operating conditions (in particular the operating temperature and time) for each plant to the plant's relative susceptibility to PWSCC. The generic responses committed to surface examinations of the VHP nozzles at the plants identified as having the highest relative susceptibility ranking. Consistent with the expectations expressed by the NRC staff in GL 97-01, the surface examinations conducted prior to November 2000 identified only limited axial cracking, and circumferential cracking below the weld in the base metal of CRDM nozzles, but no circumferential cracking above the nozzle welds and no cracking in the Alloy 182 welds.

Discussion

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. In addition, the presence of circumferential cracking at ONS3, where only a small amount of boric acid residue indicated a problem, calls into question the adequacy of current visual examinations for detecting either axial or circumferential cracking in VHP nozzles. This is especially significant if prior existing boric acid deposits on the RPV head mask the identification of new deposits. Also, the presence of insulation on the RPV head or other impediments may restrict an effective visual examination. As a remedial measure, the RPV head may have to be cleaned at a prior outage for effective identification of new deposits from VHP nozzle cracking if new deposits cannot be discriminated from existing deposits from other sources. However, the NRC staff believes that boric acid deposits that cannot be dispositioned as coming from another source should be considered, as a conservative assumption, to be from VHP nozzles, and appropriate corrective actions may be necessary. In addition, the use of special tooling or procedures may be required to provide assurance that the visual examinations will be effective in detecting the relevant conditions.

One function of VHP nozzles is to maintain the reactor coolant system pressure boundary. The CRDM nozzles support and guide the control rods, and, therefore, are relied upon in shutting down the reactor. Cracking of CRDM nozzles and welds is a degradation of the reactor coolant system boundary. Industry experience has shown that Alloy 600 is susceptible to stress corrosion cracking. 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.

The cracking identified at ONS2 and ONS3 reinforces 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. Some plants have the head insulation sufficiently offset from the RPV head to permit an effective visual examination. Other plants have the insulation offset from the head but in a contour matching that of the head, requiring special tooling and procedures to perform an effective visual examination. Still other plants have insulation directly adjacent to or attached to the RPV head, potentially requiring the removal of the insulation to permit an effective visual examination. Several licensees have recently performed expanded VT-2 examinations using remote devices to inspect between the RPV head and the insulation. 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.

For boric acid deposits from CRDM nozzle cracks to be detectable at the outer surface of the RPV head, sufficient reactor coolant has to leak through the primary pressure boundary into the annulus between the CRDM nozzle and the RPV head base metal, propagate up the annulus, and finally emerge onto the outer surface of the RPV head. Since PWSCC cracks in Alloy 600 and Alloy 182 welds are very tight, leakage from axial cracks in the nozzle and their associated welds is expected to be small. In addition, possible restraint of pressure-induced bending of circumferential cracks in CRDM nozzles could minimize the leakage available even from CRDM nozzles with large circumferential cracks, as evidenced by small boric acid deposits identified at ONS3. As described in Electric Power Research Institute (EPRI) 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" (referred to as "the MRP-44, Part 2, report"), the majority of CRDM nozzles are installed into the RPV head with an interference fit at room temperature, with 43 plants having specified interference fit ranges greater than those at ONS and ANO1. Should these interference fits persist at plant operating conditions, they could provide an impediment to the flow of coolant leakage up the annulus and thereby limit the amount of deposit available on the RPV head for detection by visual examination.

The recently identified CRDM nozzle degradation phenomena raise several issues regarding the resolution approach taken in GL 97-01:

(1) Cracking of Alloy 182 weld metal has been identified in CRDM nozzle J-groove welds for the first time. This finding raises an issue regarding the adequacy of cracking susceptibility models based only on the base metal conditions.

(2) The identification of cracking at ANO1 raises an issue regarding the adequacy of the industry's GL 97-01 susceptibility model. ANO1 cracking was predicted to be more than 15 effective full power years (EFPY) beyond January 1, 1997, from reaching the same conditions as the limiting plant, based on the susceptibility models used by the industry to address base metal cracking in response to GL 97-01.

(3) Circumferential cracking of CRDM nozzles, located outside of any structural retaining welds, has been identified for the first time. This finding raises concerns about the potential for rapidly propagating failure of CRDM nozzles and control rod ejection, causing a loss of coolant accident (LOCA).

(4) Circumferential cracking from the CRDM nozzle OD to the ID has been identified for the first time. This finding raises concerns about increased consequences of secondary effects of leakage from relatively benign axial cracks.

(5) Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective inspection methods to detect the presence of degradation in CRDM nozzles before the nozzle integrity is compromised.

After the initial finding of significant circumferential cracking at ONS3, the NRC held a public meeting with the EPRI Materials Reliability Program (MRP) on April 12, 2001, to discuss CRDM nozzle circumferential cracking issues. During the meeting, the industry representatives indicated that they were developing a generic safety assessment, recommendations for revisions of near-term inspections, and long-term inspection and flaw evaluation guidelines. On May 18, 2001, the MRP submitted the MRP-44, Part 2, report to provide an interim safety assessment for PWSCC of Alloy 600 VHP nozzles and Alloy 182 J-groove welds in PWR plants. On June 7, 2001, the NRC held a public meeting at which the MRP provided initial responses to questions on the MRP-44, Part 2, report that the NRC staff had identified and transmitted to the MRP on May 25, 2001.

The approach taken in the MRP-44, Part 2, report 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, provided in Appendix B of the MRP-44, Part 2, report, 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. To address the experience at ONS, the report recommended that plants ranked within 10 EFPY of ONS3 and having fall 2001 outages should perform a visual inspection of the RPV top head capable of detecting small amounts of leakage similar to that observed at the Oconee units and ANO1.

The NRC staff provided questions to the MRP on various aspects of the MRP-44, Part 2, report in a letter dated June 22, 2001; the MRP provided responses in a letter dated June 29, 2001. These questions addressed aspects of the proposed industry treatment that the NRC staff did not agree with. Two specific areas of concern are (1) the finding that nozzle leaks are detectable on all vessel heads, and (2) the lack of consideration of an applicable crack growth rate for the VHP nozzle cracking situation (including a conclusion in the MRP responses that the appropriate crack growth rate for OD cracking of VHP nozzles is represented by data from a primary water environment). The issue of detectibility of nozzle leaks in any particular plant is difficult to address due to a need for plant-specific as-built geometries, such as measured dimensions on CRDM nozzles and RPV penetrations to characterize the interference fit population for a particular RPV head. In addition, there is a need to provide a sufficiently detailed model of the RPV head and expected through-wall crack characteristics, such as surface roughness and crack tightness, to provide assurance that any nozzles with through-wall cracking will provide sufficient leakage to the RPV head surface such that residual deposits of boric acid will provide a detectable condition for the visual examination. An inability to provide assurance of a detectable residual deposit or to discriminate prior existing boric acid deposits caused by non-safety-significant sources from boric acid deposits caused by CRDM nozzle cracking could limit the effectiveness of visual examinations.

Because visual examination of the RPV head or volumetric examination of the VHP nozzles occurs only periodically (generally at a scheduled refueling outage), the issue of crack growth rate in VHP nozzles is an important consideration in providing assurance that VHP nozzles will maintain their structural integrity between examination opportunities. In particular, crack growth should be low enough to ensure that VHP nozzles which are determined to be unflawed during an examination do not have critical flaw sizes prior to the next scheduled examination.

From the results of the susceptibility ranking model proposed in Appendix B to MRP-44, Part 2, 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 EFPY 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 EFPY but less than 30 EFPY 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 EFPY 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 gradation of inspection effort for the subpopulations of plants noted above. Nevertheless, addressees should be cognizant of extenuating circumstances at their respective plant(s) that would suggest a need for more aggressive inspection practices to provide an appropriate level of confidence in VHP nozzle integrity. In addition, since inspection and repair activities can potentially result in large personnel exposures, licensees should ensure that all activities related to the inspection of VHP nozzles and the repair of identified degradation are planned and implemented to keep personnel exposures as low as reasonably achievable (ALARA), consistent with the NRC ALARA policy.

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.

The NRC has developed a Web page to keep the public informed of generic activities on PWR Alloy 600 weld cracking (http://www.nrc.gov/NRC/REACTOR/ALLOY-600/index.html). This page provides links to information regarding the cracking identified to date, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update this Web page as new information becomes available.

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 accomplished 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.

Requested Information

This bulletin requests addressees to submit information. Addressees who choose to utilize the analyses provided in the MRP-44, Part 2, report or similar analyses need to consider the NRC staff questions relative to this report (provided to the MRP by letter dated June 22, 2001) when preparing their plant-specific responses to the requested information. Addressees should note that the NRC staff has found that the industry response to these questions (provided by letter dated June 29, 2001) does not provide a sufficient basis for resolving the relevant technical issues and that additional information will be necessary to support the plant-specific evaluations.

Addressees are requested to provide the requested information within 30 days of the date of this bulletin (except for Item 5).

1. All addressees are requested to provide the following information:

   a. the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;

   b. a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;

   c. a description of the RPV head insulation type and configuration;

   d. a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;

   e. a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.

2. If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:

   a. a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;

   b. a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;

   c. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;

   d. your basis for concluding that the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

      (1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

      (2) If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.

3. If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:

   a. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;

   b. your basis for concluding that the inspections identified in 3.a. will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

      (1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

      (2) If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

4. If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:

   a. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;

   b. your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

      (1) If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

      (2) The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

5. Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:

   a. a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;

S14D-04675

b. if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.

Required Response

In accordance with 10 CFR 50.54(f), in order to determine whether any license should be modified, suspended, or revoked, each addressee is required to respond as described below. This information is sought to verify licensee compliance with the current licensing basis for the facilities covered by this bulletin.

Within 30 days of the date of this bulletin, each addressee is required to submit a written response indicating (1) whether the requested information will be submitted and (2) whether the requested information will be submitted within the requested time period. Addressees who choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action they propose to take, including the basis for the acceptability of the proposed alternative course of action.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50. 54(f). In addition, submit a copy of the response to the appropriate regional administrator.

Reasons for Information Request

Through-wall cracking of VHP nozzles violates NRC regulations and plant technical specifications. Circumferential cracking of VHP nozzles can pose a safety risk if permitted to progress to the point that nozzle integrity is in question and the risk of a loss of coolant accident or probability of a VHP nozzle ejection increases. This information request is necessary to permit the assessment of plant-specific compliance with NRC regulations. This information will also be used by the NRC staff to determine the need for and to guide the development of additional regulatory actions to address cracking in VHP nozzles. Such regulatory actions could include regulatory requirements for augmented inspection programs under 10 CFR 55a(g)(6)(ii) or additional generic communication.

Related Generic Communications

- Information Notice 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," April 30, 2001. [ADAMS Accession No. ML011160588]

- Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997.
- Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.
- Information Notice 90-10, "Primary Water Stress Corrosion Cracking of INCONEL 600," February 23, 1990.
- Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.
- NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this bulletin). Specifically, the requested information will enable the NRC staff to determine whether current inspection practices for the detection of cracking in the VHP nozzles at reactor facilities provide reasonable confidence that reactor coolant pressure boundary integrity is being maintained. The requested information will also enable the NRC staff to determine whether addressee inspection practices need to be augmented to ensure that the safety significance of VHP nozzle cracking remains low. No backfit is either intended or approved by the issuance of this bulletin, and the staff has not performed a backfit analysis.

Federal Register Notification

A notice of opportunity for public comment on this bulletin was not published in the *Federal Register* because the NRC staff is requesting information from power reactor licensees on an expedited basis for the purpose of assessing compliance with existing applicable regulatory requirements and the need for subsequent regulatory action. This bulletin was prompted by the discovery of circumferential cracking in CRDM nozzles (above the nozzle-to-vessel head weld) from the OD to the ID and cracking in the J-groove weld metal itself. Both of these phenomena have not been previously identified in PWRs. As the resolution of this matter progresses, the opportunity for public involvement will be provided.

S14D-04677

Paperwork Reduction Act Statement

This bulletin contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) These information collections were approved by the Office of Management and Budget, approval number 3150-0011.

The burden to the public for these mandatory information collections is 140 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or on any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

If you have any questions about this matter, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/

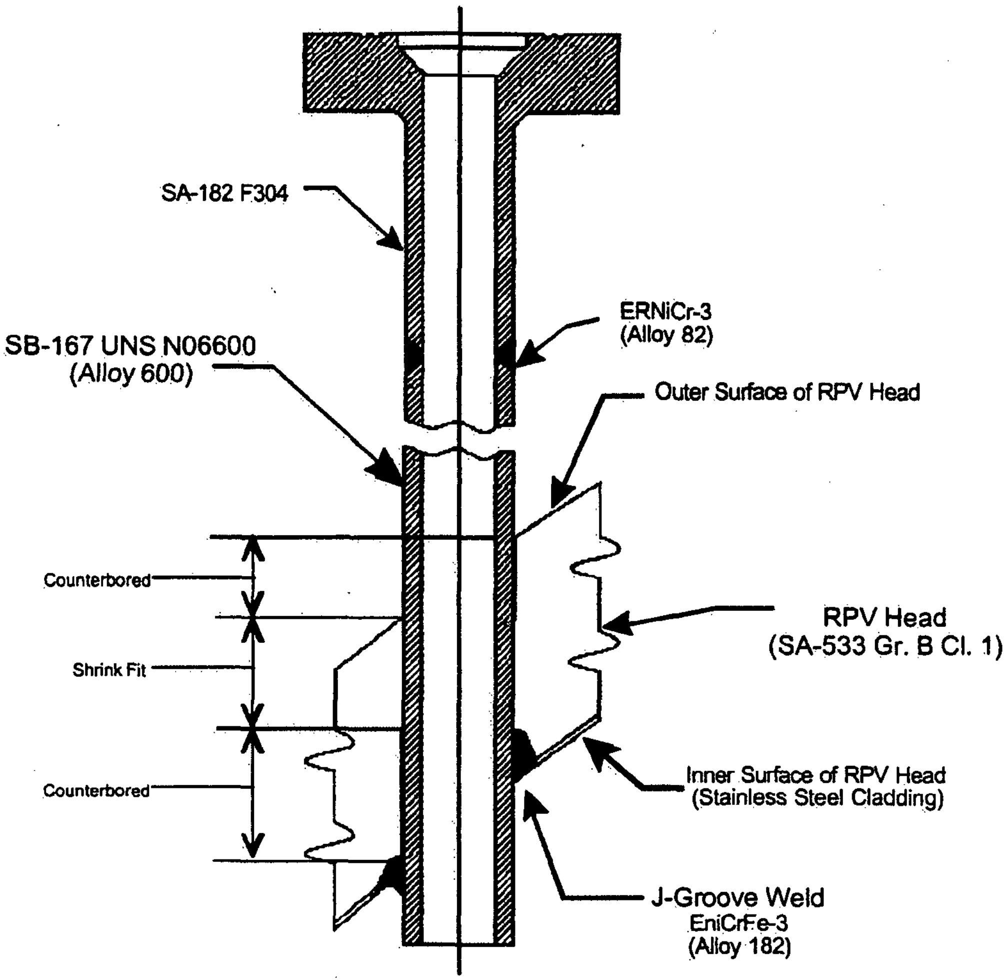
David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: Allen L. Hiser, Jr., NRR
301-415-1034
E-mail: alh1@nrc.gov

Lead Project Manager: Jacob I. Zimmerman, NRR
301-415-2426
E-mail: jiz@nrc.gov

Attachment:
Schematic Figure of Typical CRDM Nozzle Penetration

S14D-04678



Schematic Figure of Typical CRDM Nozzle Penetration

S14D-04679

RAS C-150

FirstEnergy

Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449-9760

Guy G. Campbell
Vice President - Nuclear

419-321-8588
Fax: 419-321-8337

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 9
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

September 4, 2001

Docket Number 50-346

License Number NPF-3

Serial Number 2731

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

Ladies and Gentlemen:

On August 3, 2001, the Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The Bulletin requested information regarding the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles, including the extent of nozzle leakage and cracking that has been found to date, inspections and repairs that have been completed to satisfy applicable regulatory requirements, and the basis for concluding that plans for future inspections will ensure compliance with applicable regulatory requirements.

The Davis-Besse Nuclear Power Station (DBNPS) has scheduled VHP inspections during the upcoming spring 2002 refueling outage. The FirstEnergy Nuclear Operating Company (FENOC) provides the attached information for the DBNPS in response to NRC Bulletin 2001-01.

DOCKETED
USNRC

September 9, 2009 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

NRC036-03655

TEMPLATE = SECY-028 DS 03

If you have any questions, or require further information, please contact Mr. David H. Lockwood, Manager, Regulatory Affairs, at (419) 321-8450.

Very truly yours,

RMC/s

Enclosure and Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, DB-1 NRC/NRR Project Manager
K. S. Zellers, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

RESPONSE TO

NRC BULLETIN 2001-01

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f) and contains information pursuant to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for the Davis-Besse Nuclear Power Station, Unit Number 1.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: Guy G. Campbell
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 4th day of September, 2001.

Nora L. Flood
Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

Response to NRC Bulletin 2001-01 for the Davis-Besse Nuclear Power Station

The following information is provided in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for the Davis-Besse Nuclear Power Station (DBNPS).

NRC Bulletin Request Item 1.a:

> The plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report.

Response:

The DBNPS has been analyzed for susceptibility relative to the Oconee Nuclear Station, Unit 3 (ONS3) using the Materials Reliability Program (MRP) time-at-temperature Primary Water Stress Corrosion Cracking (PWSCC) model. The parameters used in this ranking are included in Attachment 2. This evaluation showed that it will take the DBNPS 3.1 Effective Full Power Years (EFPY) of additional operation from March 1, 2001, to reach the same time-at-temperature as ONS3 when leaking nozzles were discovered in March 2001.

The DBNPS falls into the NRC category of plants within 5 EFPY of ONS3.

NRC Bulletin Request Item 1.b:

> A description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles.

Response:

The DBNPS has 69 Control Rod Drive Mechanism (CRDM) nozzles of which 61 are used for CRDMs, 7 are spare, and one is used for the Reactor Pressure Vessel (RPV) head vent piping which extends from the CRDM nozzle and terminates at the top of Steam Generator Number 2. Each CRDM nozzle is constructed of Inconel Alloy 600 and is attached to the RPV head by an Inconel Alloy 182 J-groove weld. The RPV head is constructed of carbon steel and is internally clad with stainless steel. The material for the nozzles was supplied by two suppliers. B&W Tubular Products supplied material for 60 nozzles and Huntington Alloys supplied the material for the remaining 9 nozzles. The head arrangement and requested nozzle details are provided in Attachment 2.

NRC Bulletin Request Item 1.c:

> A description of the RPV head insulation type and configuration.

Response:

The DBNPS has metal reflective horizontal vessel head insulation. Metal reflective insulation is used on the exterior of the vessel from the closure flange down to and including the exterior of the bottom head dome. Removable metal reflective insulation panels enclose the top head closure flange and studs. Metal reflective insulation is used on the RPV head. A gap exists between the RPV head and the insulation, the minimum gap being at the dome center of the RPV head where it is approximately 2 inches, and does not impede a qualified visual inspection. This is shown in the attached DBNPS drawing 7749-M-197-2-3 of the general arrangement outline for the RPV insulation.

NRC Bulletin Request Item 1.d:

> A description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations.

Response:

The DBNPS has performed two inspections within the past four years, during the 11th Refueling Outage (RFO) in April 1998 and during the 12th RFO in April 2000. The scope of the visual inspection was to inspect the bare metal RPV head area that was accessible through the weep holes to identify any boric acid leaks/deposits. The DBNPS also inspected 100% of Control Rod Drive Mechanism (CRDM) flanges for leaks in response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The results of these two recent inspections are described below.

Inspections of the RPV head are performed with the RPV head insulation installed in accordance with DBNPS procedure NG-EN-00324, "Boric Acid Corrosion Control Program," which was developed in response to Generic Letter 88-05. As stated previously, a gap exists between the RPV head and the insulation, the minimum gap being at the dome center of the RPV head where it is approximately 2 inches, and does not impede visual inspection. The service structure envelopes the DBNPS RPV head and has 18 openings (weep holes) at the bottom through which inspections are performed. There are 69 CRDM nozzles that penetrate the RPV head. The metal reflective insulation is located above the head and does not interfere with the visual inspection. The visual inspection is performed by the use of a small camera. This camera is inserted through the weep holes.

- April 1998 Inspection Results (11RFO)
  This visual inspection showed an uneven layer of boric acid deposits scattered over the head. There were some lumps of boron, with the color varying from brown to white. The outside diameter of the CRDM tubes showed white streaks, providing evidence of downward flow and attributable to CRDM flange leakage. The head was cleaned by use of a manual

scrubber and vacuum through the weepholes. The head was videotaped after cleaning for future reference.

- April 2000 Inspection Results (12RFO)
  In April 2000, Framatome Nuclear Power Services performed a 100% video inspection of CRDM flanges above the RPV insulation. Five leaking CRDM flanges were identified at locations F10, D10, C11, F8, and G9. The main source of leakage was associated with the D10 CRDM flange. Positive evidence (boron deposits on the vertical faces of the CRDM flanges and nozzle) existed that drives F8, F10 and C11 had limited gasket leakage. CRDM G9 had boron deposits under the CRDM flange between the flange and insulation, providing confidence that this leakage was associated with flange leakage. All five CRDM gaskets were replaced and the D10 CRDM flange was machined. Visual inspection of the flanges was performed. Some boric acid crystals had accumulated on the RPV head insulation beneath the leaking flanges. These deposits were cleaned (vacuumed). After cleaning, the area above the insulation was videotaped for future reference.

  Inspection of the RPV head/nozzles area indicated some accumulation of boric acid deposits. The boric acid deposits were located beneath the leaking flanges with clear evidence of downward flow. No visible evidence of nozzle leakage was detected. The RPV head area was cleaned with demineralized water to the greatest extent possible while maintaining the principles of As-Low-As-Reasonably-Achievable (ALARA) regarding the dose. Subsequent video inspection of the cleaned RPV head areas and nozzles was performed for future reference.

- Subsequent Review of 1998 and 2000 Inspection Videotapes Results
  Since May 2001, a review of the 1998 and 2000 inspection videotapes of the RPV head has been performed. This review was conducted to re-confirm the indications of boron leakage experienced at the DBNPS were not similar to the indications seen at ONS and ANO-1; i.e., was not indicative of RPV nozzle leakage. This review determined that indications such as those that would result from RPV head penetration leakage were not evident.

NRC Bulletin Request Item 1.e:

A description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.

Response:

The lower section of the service structure is welded to the head. The service structure then bolts to this lower section. Fan holes are provided to allow forced air cooling of CRDMs. Ductwork connected to two remotely mounted, 100 percent capacity cooling fans is mounted over the fan holes in the service structure. The lower portion of the service structure is also provided with ledges to support the RPV head insulation. The upper portion of the service structure cylinder is provided with a monorail to accommodate chain hoists that are required for stud tensioner handling. A deck is provided on the service structure to provide a work platform for servicing the CRDMs. This deck also provides the support for the CRDM cooling water manifolds and electrical cables. The deck is composed of individual butted plates with openings to accept seismic clamps provided with the CRDMs. These seismic plates provide stability for the upper portion of the CRDM. They are field-aligned to the reactor vessel control rod nozzles

Additional components that are located above the RVP head and below the missile shield within the refueling canal include the RPV head vent line piping, CRDM cabling, cooling water piping for CRDM thermal barriers, and miscellaneous electrical power cables.

The elevations for the Reactor Coolant System (RCS), including the top of the CRD Closure Housings at the top of the service structure, are shown in the attached Figure 1 and Figure 2. The top of the missile shield over the service structure is at elevation 653'0". The missile shield is comprised of six concrete removable panels, each 31' 5" x 6' 6" x 3'. It spans the refueling canal and is supported on both sides by the Steam Generator "D-Ring" walls.

NRC Bulletin Request Item 2:

> If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information: [a, b, c, d]

Response:

The DBNPS has not previously identified either leakage from or cracking of its RPV head penetration nozzles.

NRC Bulletin Request Item 3.a:

> If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:
>
> a. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule.

Response:

The DBNPS plans for future inspections consist of the following:

1. A qualified visual examination of the RPV head will be performed during 13RFO, which is currently scheduled for April 2002.

Visual examinations have been performed during each refueling outage and reviewed by the engineering staff. For the 13RFO, a qualified visual examination will be performed. Personnel performing this task will be instructed on the type of unacceptable conditions using ONS3 as the basis. Inspections will be performed in accordance with a procedure developed specifically for these examinations that will meet the basic requirements of an ASME VT-2 inspection, and will not be compromised due to any pre-existing boric acid crystal deposits. The previous inspection video of the cleaned head and flanges will be used to help determine any unacceptable conditions. The RPV head will be cleaned (as necessary) and videotaped prior to return to service to re-establish a baseline for future inspections.

The acceptance criteria to be used will consist of comparative evaluations of any as-found boric acid crystal deposits to photographs of leaking CRDM nozzles observed at ONS3 and Arkansas Nuclear One-Unit 1 (ANO-1) and evaluation against any identified leaking CRDM nozzle flanges. The cracks leading to the leak will be characterized by supplemental examination and the nozzle will be repaired.

Because there are significant efforts being undertaken by the MRP and the nuclear industry to better understand this phenomena and to develop optimized inspections methods (including tooling), mitigation and repair techniques, the foregoing is an interim response to NRC Bulletin Request 3.a reflecting the current plans based on information currently available. The FirstEnergy Nuclear Operating Company (FENOC) proposes to provide a final response to NRC Bulletin Request 3.a by January 29, 2002 (60 days before the start of 13RFO scheduled for the spring of 2002). Final plans will be based on the inspection results from other facilities, the ongoing work of the MRP, and the advancement of Non-Destructive Examination (NDE) technology and development of remote tooling adequate to perform effective and timely surface or volumetric examinations from underneath the RVP head.

A flow chart of the inspection plan is shown in Figure 3. Details of the inspection plan will be developed prior to the 13RFO.

2. Qualified visual examinations will continue to be performed at subsequent refueling outages.

The DBNPS will continue to perform qualified visual examinations of the RPV head for evidence of leaking CRDM nozzles at subsequent refueling outages. The visual examination procedure will be updated, as required, to include industry experience.

<u>NRC Bulletin Request Item 3.b:</u>

Your basis for concluding that the inspections identified in 3.a. will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

(1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in

the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

(2) If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

Response:

The DBNPS is similar in design to ONS3 and ANO-1, which have demonstrated an ability to identify leaking CRDM nozzles by visual inspection for boric acid crystal deposits. This has been demonstrated at these units by examination of additional non-leaking nozzles for signs of cracking. In each of the twenty-six nozzles, the results did not find any signs of significant cracking, thereby providing the necessary confidence that leaking CRDM nozzles can be found by visual inspection. The DBNPS fabrication records were reviewed to determine how CRDM bores were machined and how CRDM nozzles were installed. CRDM nozzles were installed in the RPV closure head with a designed 0.0005 inches to 0.0015 inches of diametral interference (documented in "Safety Evaluation for B&W-Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking," BAW-10190P, dated May 1993). The CRDM nozzle shaft diameter is custom ground to 0.001 inches greater than the final diameter of the associated CRDM bore with a 32AA finish. A general description of the CRDM bores machining is as follows:

- Rough machine CRDM bores (Note: DBNPS RPV head penetrations were not counterbored.)
- Final heat treatment of RV closure head
- Finish machine CRDM bores to a 250 finish

A general description of the CRDM nozzle installation is as follows:

- Cool CRDM nozzles in liquid nitrogen to -140°F minimum
- Install CRDM nozzle in specified location
- Allow CRDM nozzle to warm to 70°F

During the final Quality Assurance inspection, CRDM bores were inspected for final top and bottom bore diameter and verticality. After individual CRDM nozzle shaft custom grinding to approximately 0.001 inches greater in diameter than the final CRDM bore diameter, CRDM nozzle shafts were also measured at both the top and the bottom of the custom ground length. CRDM nozzle shafts are longer than CRDM bores are deep. Thus, CRDM nozzle shaft diameter measurements do not directly line up with CRDM bore diameter measurements, although in the case of the DBNPS these locations should be fairly close because of the lack of counterbores. Therefore, the resulting top and bottom dimensional fits are considered approximate. The values for the DBNPS RPV head are calculated to range from a maximum interference fit of 0.0021 inches to a gap of 0.0010 inches.

In 1993, the B&WOG performed a safety evaluation for CRDM nozzle cracking (reference: previously cited BAW-10190P). In this evaluation, a 3D finite element model of all major components of a hillside CRDM nozzle-to-head welded structure was constructed. The B&WOG calculation includes the maximum 0.010 inch diametric counterbore at the top and bottom locations (typical for most B&WOG plant designs), which tends to increase the stresses in the nozzle and is bounding for the DBNPS. During operation, an interference fit is calculated to release to become a gap due to temperature and pressure dilation, which provides a leak path for a through-wall crack that allows detection by visual inspection. The B&WOG calculation assumes a nominal 0.001 inch interference fit, which will open to a maximum gap of 0.0033 inches during operation.

As noted earlier, leakage from this gap has been demonstrated at both ONS and ANO-1, for which interference fits of up to 0.0014 inches have been calculated from the final QA inspection data (as documented in MRP-44, Part 2). Figure 4 provides a graphical representation of these data. The largest interference fit at the DBNPS occurs on nozzle number 50 which, as stated previously, has been calculated at 0.0021 inches at the top. This same nozzle also has an interference fit of 0.0010 inches at the bottom. Thus, the 0.0033 inch gap during operation would be somewhat less for the DBNPS, assuming the 0.0021 inch interference fit (instead of the nominal 0.001 inch). This gap would still be expected to provide a leak path to the top of the RPV head in the event of a cracked CRDM nozzle or J-groove weld. The DBNPS has not observed any leakage from these paths during its past inspection activities.

The DBNPS plans to perform inspections of the RPV head and CRDM nozzles as recommended by MRP-48. The inspections will consist of qualified visual inspections of the top RPV head bare metal surface at the 13RFO scheduled for the spring of 2002. If any leaks are detected, the source will be determined, the cracks leading to the leak will be characterized by supplemental examination and the nozzle will be repaired.

As stated previously, because there are significant efforts being undertaken by the MRP and the nuclear industry to better understand this phenomena and to develop optimized inspections methods (including tooling), mitigation and repair techniques, the foregoing is an interim response reflecting the current plans based on information currently available. The FENOC proposes to provide a final response by January 29, 2002 (60 days before the start of 13RFO scheduled for the spring of 2002). Final plans will be based on the inspection results from other facilities, the ongoing work of the MRP, and the advancement of NDE technology and development of remote tooling adequate to perform effective and timely surface or volumetric examinations from underneath the RVP head.

The Applicable Regulatory Requirements section of the Bulletin lists the following regulatory requirements and plant commitments as providing the basis for the Bulletin assessment:

- Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants"

  Criterion 14 – "Reactor Coolant Pressure Boundary"

Criterion 31 – "Fracture Prevention of Reactor Coolant Boundary," and

Criterion 32 – "Inspection of Reactor Coolant Pressure Boundary"

- Plant Technical Specifications
- 10 CFR 50.55a, Codes and Standards, which incorporates by reference Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code"
- Appendix B of 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria V, "Instructions, Procedures, and Drawings;" IX, "Control of Special Processes;" and XVI, "Corrective Actions"

The following addresses each of these criteria and demonstrates that the criteria will be met for the DBNPS until the inspections are performed.

<u>Design Requirements: 10 CFR 50, Appendix A – General Design Requirements</u>

The Bulletin states:

"The applicable GDC [General Design Criteria] 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."
These referenced criteria state the following:

- Criterion 14 – Reactor Coolant Pressure Boundary
  "The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."
- Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary
  "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance,

testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws."

- Criterion 32 – Inspection of Reactor Coolant Pressure Boundary
  "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."

During initial licensing of the DBNPS it was demonstrated that the design of the reactor coolant pressure boundary met the requirements in place at that time. The GDC included in Appendix A to 10 CFR 50 did not become effective until May 21, 1971. The construction permit for the DBNPS was issued prior to May 21, 1971; consequently, the DBNPS was not subject to the GDC requirements (reference: SECY-92-223; 9/18/92). However, the following demonstrates compliance with the design criteria for the RPV head nozzles.

- Pressurized water reactors licensed both before and after issuance of Appendix A to Part 50 (1971) complied with these criteria in part by: 1) selecting Alloy 600, and other austenitic materials with excellent corrosion resistance and extremely high fracture toughness, for reactor coolant pressure boundary materials, and 2) following ASME Codes and Standards and other applicable requirements for fabrication, erection, and testing of the pressure boundary parts. NRC reviews of operating license submittals subsequent to issuance of Appendix A included evaluating designs for compliance with the General Design Criteria.
- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the RPV top head nozzles at some plants. The suitability of the originally selected materials has been confirmed. The robustness of the design has been demonstrated by the small amounts of the leakage that has occurred and by the fact that none of the cracks in Alloy 600 reactor coolant pressure boundary materials has rapidly propagated or resulted in catastrophic failure or gross rupture. Given the inherently high fracture toughness and flaw tolerance of the Alloy 600 material there is indeed an extremely low probability of a rapidly propagating failure and gross rupture. It should be noted that the originally proposed Appendix A (July 1967) was written in terms of extremely low probability of gross rupture or <u>significant</u> leakage throughout the design life.
- Utilizing the conservative time-at-temperature ranking model of MRP-44, the operating time before Davis-Besse would reach an equivalent degradation time as ONS-3 is at least 3.1 EFPY.
- An updated safety assessment was performed by Framatome-ANP in April 2001 to address the CRDM nozzle cracking observed at ONS-1, ONS-3, and ANO-1. Flaw growth calculations were performed, using the modified Peter Scott crack growth equation and assuming an initial flaw length of 180° around the nozzle, which indicate that it would take approximately 4 years for a through-wall flaw to grow another 25% around the circumference. This remaining ligament, which would be 25% of the original circumference,

would still be sufficient to preclude gross net-section failure (nozzle ejection). This ligament satisfies primary stress limits using a safety factor of 3.

- The revised Framatome ANP safety assessment of April 2001 also concluded that simultaneous multiple CRDM nozzles will not fail and that the failure of a single CRDM nozzle is bounded by both the LOCA and non-LOCA plant analyses already completed to support current plant operation.
- MRP-44, Appendix C describes the accident sequence analyses already in place using the DBNPS Emergency Operating Procedures (EOPs). The existing EOPs provide adequate directions to mitigate any transient that would occur should there be a failure of a CRDM nozzle.
- All evidence to date suggests that it will require several years for the material to degrade to the point that total failure of the component could occur. During that time, if a crack should form, leakage of primary coolant on the RPV head can be identified through routine visual inspection of the bare RPV head. The component can then be repaired and returned to service without jeopardizing the health and safety of the public.

Therefore, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32, respectively, were satisfied during the initial licensing review for the DBNPS, and continue to be satisfied during operation even in the presence of a potential for stress corrosion cracking of the RPV head penetration nozzles.

<u>Operating Requirement: 10 CFR 50.36 - Technical Specifications</u>

The Bulletin states:

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

10CFR 50.36 contains requirements for Plant Technical Specifications. Paragraphs 2 and 3 of 10 CFR 50.36 are particularly relevant:

- 10CFR 50.36(c)(2) Limiting Conditions for Operation

  "(i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

  (ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

  (C) Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

- 10 CFR 50.36(c)(3) Surveillance Requirements

  "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

The reactor coolant pressure boundary provides one of the critical barriers that guard against the uncontrolled release of radioactivity. Therefore, the DBNPS Technical Specification 3.4.6.2 includes a requirement and associated action statements addressing reactor coolant pressure boundary leakage. The limits for reactor coolant pressure boundary leakage are stated in terms of the amount of leakage, *e.g.*, 1 gallon per minute for unidentified leakage; ≤10 gpm for identified leakage; and no reactor coolant system pressure boundary leakage.

Leaks from Alloy 600 RVP head penetrations due to PWSCC have been well below the sensitivity of on-line leakage detection systems. Plants have evaluated this condition and have determined that the appropriate inspections are bare-metal visual inspections for boric acid deposits during plant shutdowns. If leakage or unacceptable indications are found, the defect must be repaired before the plant goes back on line. If through-wall boundary leaks of the CRDM nozzles increase to the point where they are detected by the on-line leak detection systems, then the leak must be evaluated per the Technical Specification's specified acceptance criteria and the Technical Specification's required actions taken.

Inspection Requirements: 10 CFR 50.55a and ASME Section XI

The Bulletin states:

"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 [IWB-2500-1[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."

[1] An erratum appears to exist in the Bulletin. Table IWA-2500-1 is cited, but does not exist. It appears the citation should have been IWB-2500-1.

"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."

10 CFR 50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components." Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

The DBNPS performs visual inspections for evidence of leakage by examining the RPV head surface and the CRDM flanges per the requirements of NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." If pressure boundary leakage is suspected, supplemental examinations of the affected CRDM nozzle will be performed to characterize the integrity of the nozzle. Some plants have conducted inspections beyond those required by Section XI and NRC Generic Letter 88-05. These inspections have included visual examinations of 100% of the bare metal surfaces of the RPV head; eddy current and liquid penetrant surface examinations; and supplemental examinations of the nozzles. These supplemental inspections coupled with the evaluations of cracking that has been found are considered to have provided a defense-in-depth approach for investigating and resolving this issue.

The acceptance standards are as detailed in Technical Specifications for pressure boundary leakage since the program under Generic Letter 88-05 is not a Code-required inspection program.

Flaws identified by supplemental methods will be evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. Any flaw not meeting the requirements for the intended service period would be repaired prior to returning it to service.

Repairs to RPV head nozzles will be performed in accordance with Section XI requirements, NRC-approved ASME Code Case requirements, or an alternative repair or replacement method approved by the NRC.

The DBNPS complies with these ASME Code requirements through implementation of the Inservice Inspection Program. In addition, additional inspections are conducted in accordance with the program developed to meet Generic Letter 88-05. If a VT-2 or qualified visual examination detects the cracks or leakage in the CRDM nozzles, corrective actions will be performed in accordance with the DBNPS corrective action program. No new plant actions are necessary to satisfy the regulatory criteria.

Quality Assurance Requirements: 10 CFR 50, Appendix B

The Bulletin states:

"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."

The design shrink fit of the CRDM nozzles at the DBNPS is similar to the design shrink fit of the ONS units indicating that through wall cracking of the nozzles of the magnitude seen at ONS should produce visually detectable evidence of leakage on the RPV head. The qualified visual inspection and the personnel involved in the evaluation of the results will be VT-2 qualified and familiar with the anticipated type of indication that any leakage would cause. Any other NDE techniques and associated equipment that may be required is presently being developed and should be qualified for the DBNPS 13RFO in the spring of 2002.

The Bulletin further states:

"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 accomplished 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."

The efforts undertaken to inspect, evaluate, and /or repair the DBNPS RPV head penetrations will be conducted and documented in accordance with procedures which comply with the FENOC Quality Assurance Program and Criterion V of 10 CFR 50, Appendix B.

The final criterion cited by the Bulletin is stated as follows:

"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."

In addressing Criterion XVI, there are two important attributes pertinent to RPV CRDM nozzles cracking.

First, Criterion XVI states "Measures shall be established to assure that conditions adverse to quality...are promptly identified and corrected." This criterion is partially met by the DBNPS's awareness of industry experience, and has been implemented in this manner in the DBNPS corrective action program whereby industry experience is evaluated for applicability to DBNPS and the applicable corrective actions, as needed are determined. This is consistent with the NRC's generic communication process, implemented by Information Notices, which reports industry experience, but does not require a response to NRC. Licensees are expected to evaluate the applicability of the information contained in the Information Notice and document a specific assessment for possible NRC review.

Criterion XVI provides the objectives and goals of the corrective action program, but leaves to the licensee the responsibility for determining the specific process to accomplish these objectives and goals. With regard to the Bulletin response, Criterion XVI does not provide specific guidance as to what is an appropriate response, but rather, the licensee is responsible for determining actions necessary to maintain public health and safety. In this particular instance, the licensee must justify its actions for addressing the PWSCC of RPV head nozzles. Furthermore, the regulatory criteria of 10 CFR 50.109(a)(7) provides supporting evidence when it states "...if there are two or more ways to achieve compliance...then ordinarily the applicant or licensee is free to choose the way which best suits its purposes."

The second attribute of Criterion XVI stated is "In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." The Bulletin suggests that for RPV head nozzle cracking, the root cause determination is important to understanding the nature of the degradation and the required actions to mitigate future cracking. As part of the DBNPS corrective action program, determination of the cause of the PWSCC in the RPV head nozzles, either through the DBNPS's efforts or as part of an industry effort, would be performed, if cracks are detected.

In summary, the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking is in compliance with the performance-based objectives of 10 CFR 50, Appendix B.

NRC Bulletin Request Item 4:

If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information: [a and b]

Response:

This request does not apply to the DBNPS because the DBNPS susceptibility ranking is within 3.1 EFPY of ONS3.

NRC Bulletin Request 5:

Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:

a. a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;
b. if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.

Response:

The DBNPS will provide the NRC with the following information within 30 days after plant restart following the 13th RFO scheduled to begin in the spring of 2002:

a. A description of the extent of RPV head nozzle leakage and cracking. This information will include the number, location, size and nature of each crack detected.

b. A description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs and other corrective actions taken to satisfy applicable regulatory requirements.

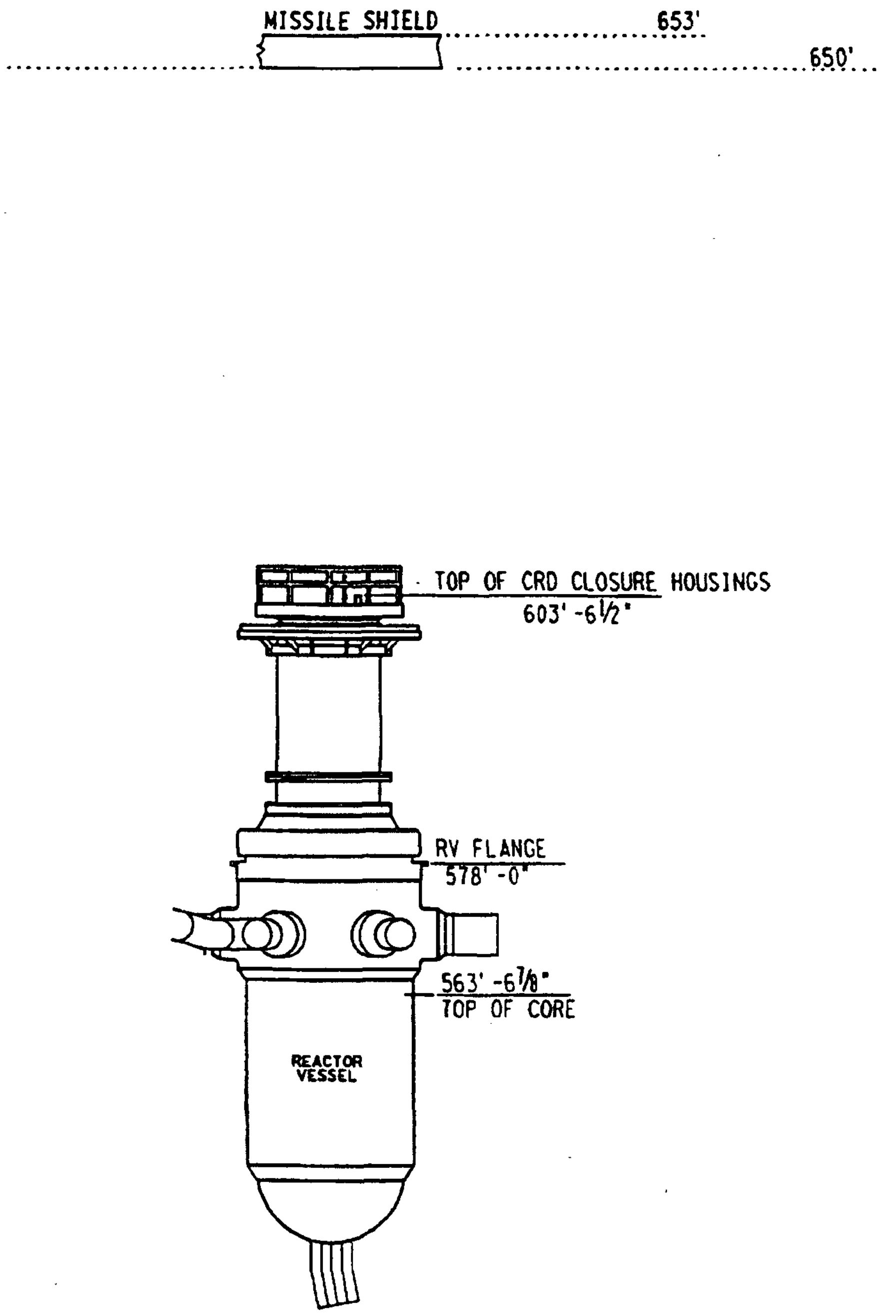

MISSILE SHIELD
653'
650'
TOP OF CRD CLOSURE HOUSINGS
603' -6½"
RV FLANGE
578' -0"
563' -6⅞"
TOP OF CORE
REACTOR
VESSEL

Figure 1

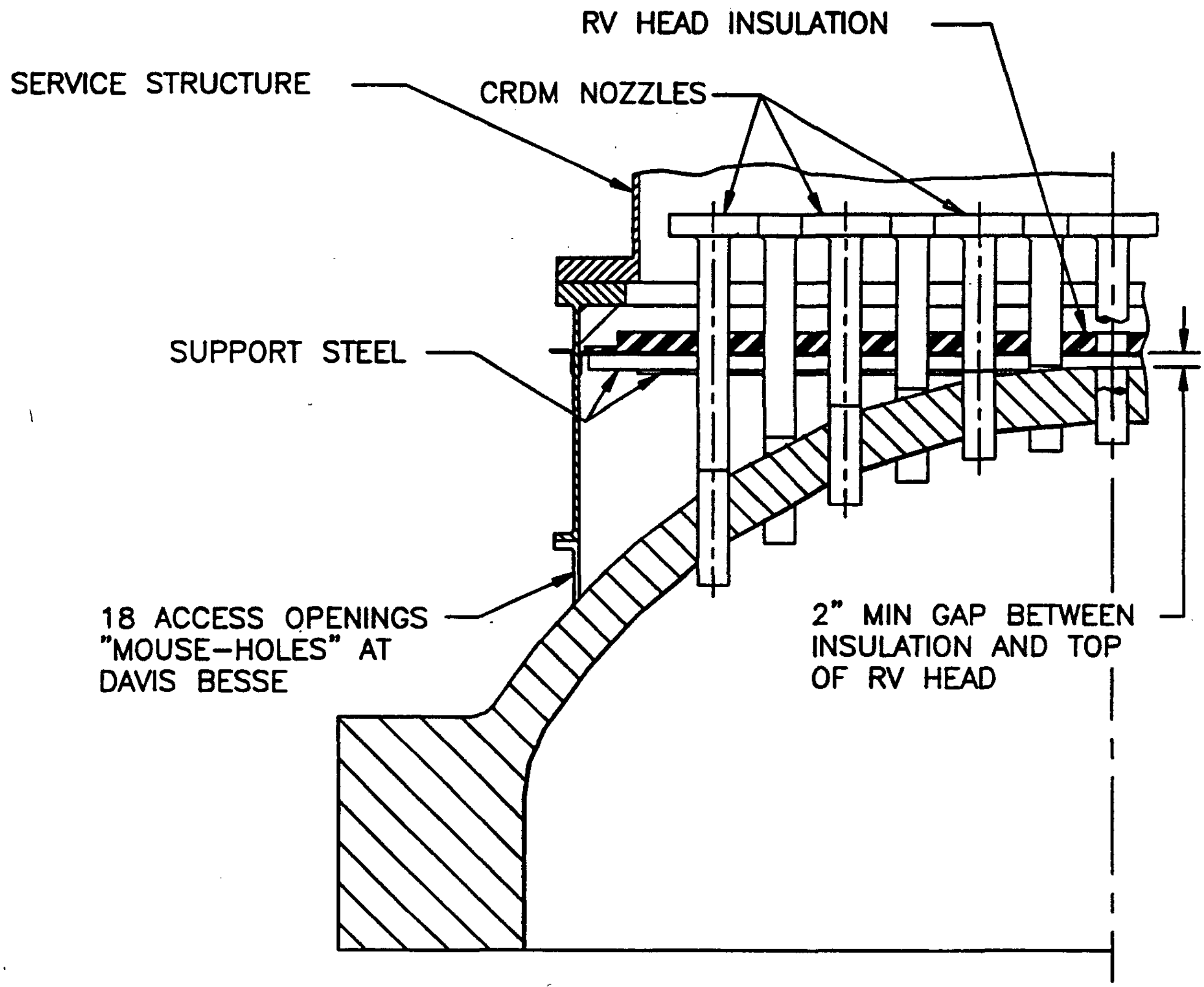


Figure 2.
Side View Schematic of Davis-Besse Reactor Vessel Head, CRDM Nozzles, and Insulation.

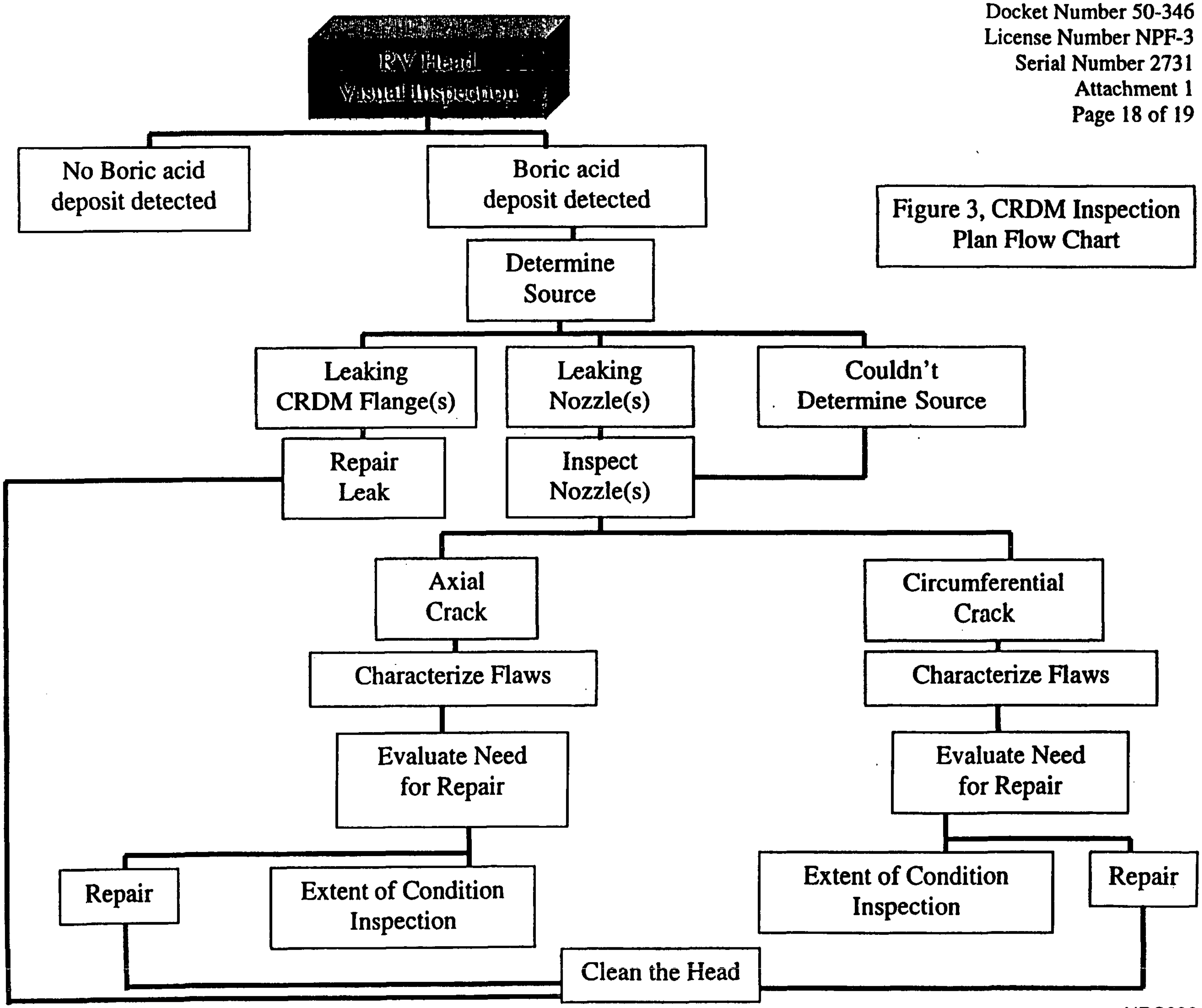
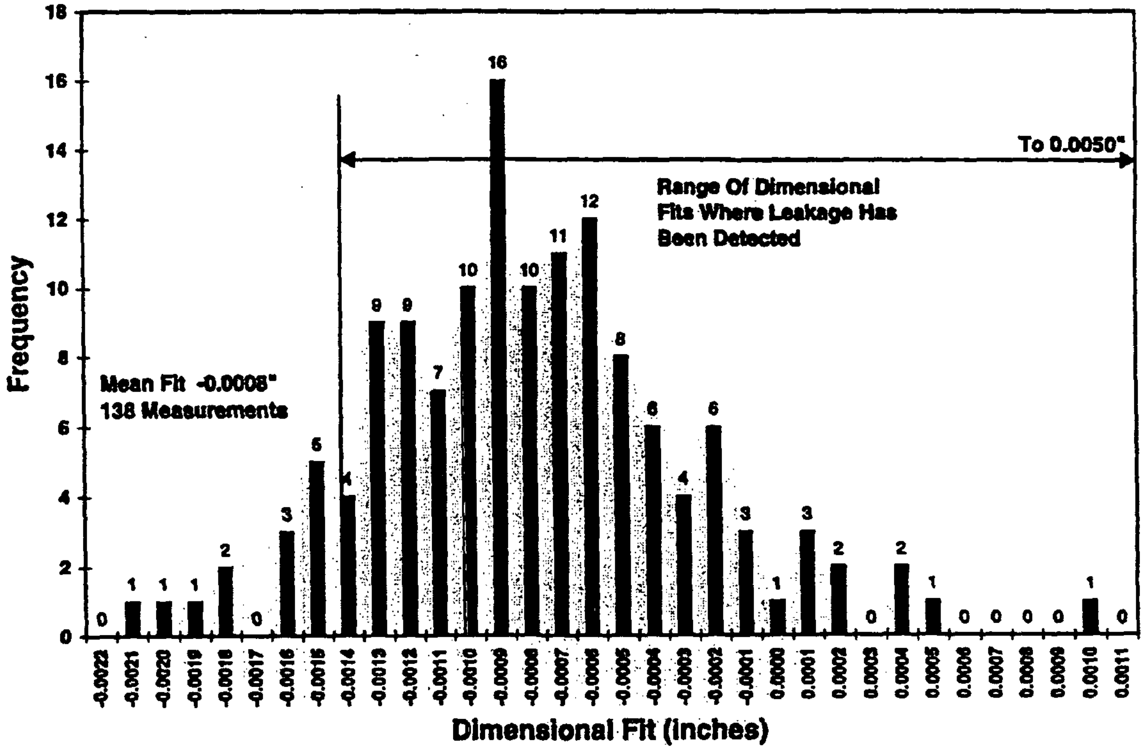


Figure 3, CRDM Inspection Plan Flow Chart

Figure 4



Distribution of Dimensional Fits in DBNPS RPV Head

Key Information

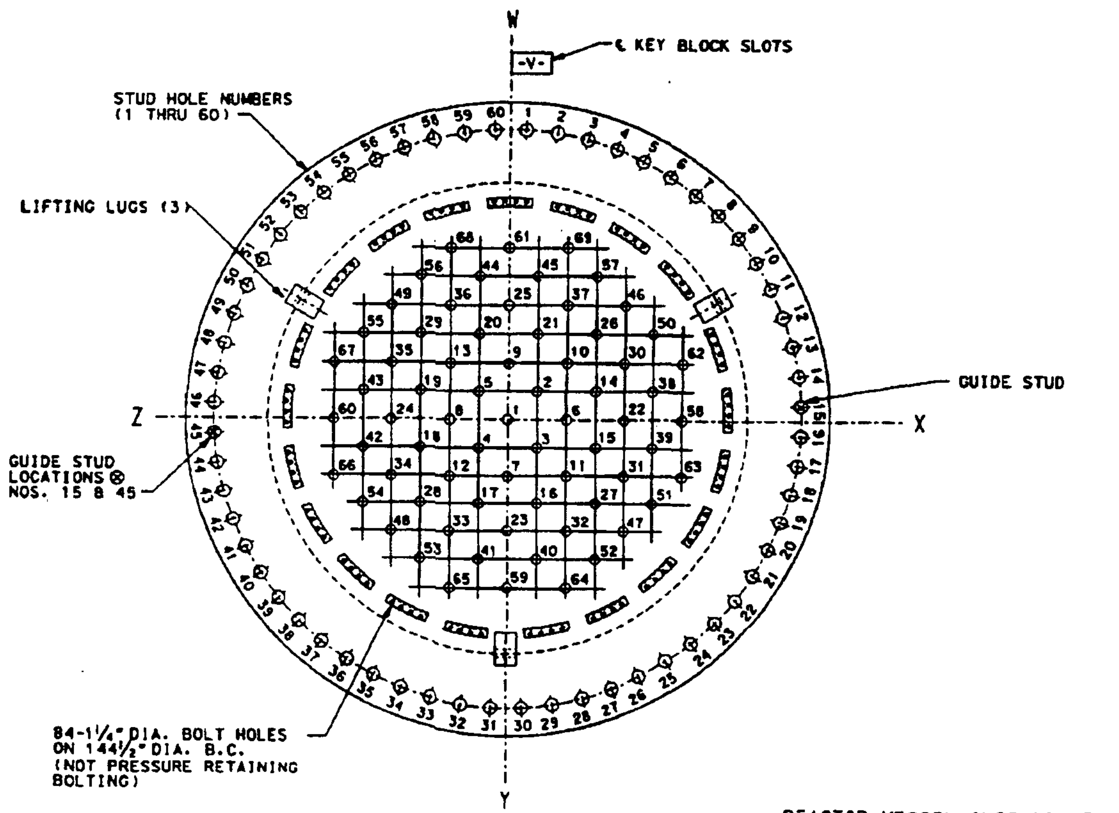
- EPRI Susceptibility Determination Criteria:

Davis-Besse Nuclear Power Station	
Lifetime EFPY	14.7
RPV Head Temperature	605.0°F

- CRDM Nozzle Information

Number of Nozzles	69
Inside Diameter	2.765 in.
Outside Diameter	4.001 in.
Minimum Distance Between Nozzle Centerlines	12.15 in.

- Nozzle Arrangement – 69 CRDM Nozzles



KEY PLAN

REACTOR VESSEL CLOSURE HEAD BOLTING DETAILS FIGURE

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs ((419)-321-8450)) at the DBNPS of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
For the 13RFO, a qualified visual examination will be performed. Personnel performing this task will be instructed on the type of unacceptable conditions using ONS3 as the basis. Inspections will be performed in accordance with a procedure developed specifically for these examinations that will meet the basic requirements of an ASME VT-2 inspection. The previous inspection video of the cleaned head will be used to help determine any unacceptable conditions. The RPV head will be cleaned (as necessary) and videotaped prior to return to service to re-establish a baseline for future inspections.	13th RFO
The acceptance criteria to be used for the qualified visual inspection will consist of comparative evaluations of any as-found boric acid crystal deposits to photographs of leaking CRDM nozzles observed at ONS3 and Arkansas Nuclear One-Unit 1 (ANO-1) and evaluation against any identified leaking CRDM nozzle flanges.	13th RFO
The DBNPS plans to perform inspections of the RPV head and CRDM nozzles as recommended by MRP-48. The inspections will consist of qualified visual inspections of the top RPV head bare metal surface at the 13th RFO scheduled for the spring of 2002. If any leaks are detected, the source will be determined, the cracks leading to the leak characterized by supplemental examination and the nozzle will be repaired.	13th RFO
The FirstEnergy Nuclear Operating Company (FENOC) proposes to provide a final response to NRC Bulletin Request 3.a by January 29, 2002 (60 days before the start of 13RFO scheduled for the spring of 2002). Final plans will be based on the inspection results from other facilities, the ongoing work of the MRP, and the advancement of Non-Destructive Examination (NDE) technology and development of remote tooling adequate to perform effective and timely surface or volumetric examinations from underneath the RVP head.	January 29, 2002

COMMITMENT LIST (continued)

COMMITMENTS	DUE DATE
Details of the qualified inspection plan will be developed prior to the 13RFO.	March 30, 2002
The DBNPS will continue to perform qualified visual examinations of the RPV head for evidence of leaking CRDM nozzles at subsequent refueling outages. The visual examination procedure will be updated, as required, to include industry experience.	Ongoing
Flaws identified by NDE methods during the CRDM nozzles inspections that are beyond current requirements will be evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code	Ongoing
The DBNPS will provide the NRC with the following information within 30 days after plant restart following the 13th RFO scheduled to begin in the spring of 2002: a. A description of the extent of RPV head nozzle leakage and cracking, if detected. This information will include the number, location, size and nature of each crack detected. b. A description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs and other corrective actions taken to satisfy applicable regulatory requirements.	30 days following end of 13th RFO

Drawing 7749-M-197-2-3 is an attachment to this letter. Drawing is already in Records Management.

Regulatory Affairs has a copy of the above drawing.

RAS C-151

(1) RECO RA

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 10
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

DOCKETED
USNRC
September 9, 2009 (11:00am)
OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

(3) SUMMARY (Log No., Title Subject)
Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reac

(4) COMMITMENT LIST ADDED TO LETTER ☒

(5) PERIC ☐ YES

(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC
Target Date 8/30/01; Required 9/4/01 ☐ N/A

(7) SPECI ☒ EXP

(8) PREPARED BY
Rod Cook ext. 7782

(9) NOTAI ☒ YES

(11) ADDITIONAL REFERENCES

(13) COMMENTS

Approvals continued on Page 2

Consisten with industry & 45 inspection expansion & later 60 days

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ COGNIZANT REGULATORY AFFAIRS INDIVIDUAL	R.M. Cook	RMC	8/20/01	8/21/01
☒ RESPONSIBLE ENGINEER – MECHANICAL DESIGN	P. Goyal	P Goyal	8/28/01	8/28/01
☒ RESPONSIBLE SUPERVISOR – MECH/STR DESIGN	T. Swim	TS	8/28/01	8/28/01
☒ RESPONSIBLE ENGINEER – PLANT ENGINEERING	A.J. Siemaszko	See attached		
☒ RESPONSIBLE SUPERVISOR – PLANT ENGRG MECH	J.B. Cunnings	See attached		
☒ RESPONSIBLE WORK CONTROL INDIVIDUAL	M. McLaughlin	See attached		
~~☒ RESPONSIBLE WORK PROD SUPDT~~	~~D.E. Missig~~	NA		
~~☒ RESPONSIBLE LICENSING INDIVIDUAL~~	~~F.W. Kennedy~~	NA		
☒ SUPERVISOR, DB LICENSING	D.R. Wuokko	Per telecon 8/28/01	8/27/01	8/28/01
☒ SUPERVISOR, DB COMPLIANCE	D.L. Miller	DLM	8/30/01	9/30/01
☒ MANAGER, REGULATORY AFFAIRS	D. H. Lockwood	DHL	8-30-01	9-4-01
☒ VICE PRESIDENT	G.G. Campbell	GG	8/23/01	9-4-01

DATE ADDED TO LETTER 9/4/01
(15) DATE ADDED BY Diana Chambers
DATE SENT TO NRC 9/4/01
(16) DISTRIBUTED BY Diana Chambers
DATE OF BLIND DISTRIBUTION 9/6/01
(18) DISTRIBUTED BY Diana Chambers

(17) ADDITIONAL DISTRIBUTION
1) Roger Huston - LSS
2) e-mail to koc@nei.org (Kurt Cozens)
3) John M. Maracek
4) R.G. Donne
5) BWOG

S11-00592

TEMPLATE = 028

DS 02

NRC LETTERS - REVIEW AND APPROVAL REPORT
ED 7159-7

(1) RECORDS MANAGEMENT NO.	(2) SERIAL NO.
	2731 Page 2

(3) SUMMARY (Log No., Title Subject)
Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

(4) COMMITMENT LIST ADDED TO LETTER ☒	(5) PERIODIC / NON-PERIODIC REPORT ☐ YES ☒ NO REPORT NO.____	
(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC Target Date 8/30/01; Required 9/4/01 ☐ N/A	7) SPECIAL HANDLING ☐ EXPRESS DELIVERY ☒ TELECOPY	DATE SENT
(8) PREPARED BY Rod Cook ext 7782	(9) NOTARY ☒ YES ☐ NO	(10) LICENSE FEE REQUIRED ☐ YES ☒ NO

(11) ADDITIONAL REFERENCES	(12) COMMITMENT NO.(S) CLOSED

(13) COMMENTS

See Page 1

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ DESIGN ENGINEERING MANAGER	D.C. Geisen			
☒ PLANT ENGINEERING MANAGER	D.L. Eshelman			
☒ WORK CONTROL MANAGER	C.D. Nelson	[signature] FOR C.D. NELSON	8-28-01	8-28-01
☒ DIRECTOR, WORK MANAGEMENT	J. Messina	JM	8-28-01	8-29-01
☒ DIRECTOR, TECHNICAL SERVICES	S.P. Moffitt			
☒ DIRECTOR, NUCLEAR SERVICES	L.W. Worley			
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DATE ADDED TO LETTER ☒	(15) DATE ADDED BY	(17) ADDITIONAL DISTRIBUTION
DATE SENT TO NRC	(16) DISTRIBUTED BY	
DATE OF BLIND DISTRIBUTION	(16) DISTRIBUTED BY	

See page 1

S11-00594

NRC027-1694

NRC LETTERS - REVIEW AND APPROVAL REPORT

ED 7159-7

(1) RECORDS MANAGEMENT NO.	(2) SERIAL NO. 2731 Page 1

(3) SUMMARY (Log No., Title Subject)
Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

(4) COMMITMENT LIST ADDED TO LETTER ☒	(5) PERIODIC / NON-PERIODIC REPORT ☐ YES ☒ NO REPORT NO.

(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC Target Date 8/30/01; Required 9/4/01 ☐ N/A	7) SPECIAL HANDLING ☐ EXPRESS DELIVERY ☒ TELECOPY	DATE SENT
(8) PREPARED BY Rod Cook ext. 7782	(9) NOTARY ☒ YES ☐ NO	(10) LICENSE FEE REQUIRED ☐ YES ☒ NO

(11) ADDITIONA

(12) COMMITMENT NO.(S) CLOSED

Plant Engrg
Green Sheet in
Moffitt's Office

(13) COMMENT

Approvals c

(14)		INITIALS	DATE RECEIVED	DATE APPROVED
☒ COGN	R.M. Cook	RMC	8/20/01	8/21/01
☒ RESPONSIBLE ENGINEER – MECHANICAL DESIGN	P. Goyal			
☒ RESPONSIBLE SUPERVISOR – MECH/STR DESIGN	T. Swim			
☒ RESPONSIBLE ENGINEER – PLANT ENGINEERING	A.J. Siemaszko			
☒ RESPONSIBLE SUPERVISOR – PLANT ENGRG MECH	J.B. Cunnings			
☒ RESPONSIBLE WORK CONTROL INDIVIDUAL	M. McLaughlin	MM	8/29/01	8/29/01
☒ RESPONSIBLE WORK PROD SUPDT	D.E. Missig			
☒ RESPONSIBLE LICENSING INDIVIDUAL	F.W. Kennedy			
☐ SUPERVISOR, DB LICENSING	D.R. Wuokko			
☐ SUPERVISOR, DB COMPLIANCE	D.L. Miller			
☒ MANAGER, REGULATORY AFFAIRS	D. H. Lockwood			
☒ VICE PRESIDENT	G.G. Campbell			

DATE ADDED TO LETTER ☐	(15) DATE ADDED BY	(17) ADDITIONAL DISTRIBUTION
DATE SENT TO NRC	(16) DISTRIBUTED BY	
DATE OF BLIND DISTRIBUTION	(18) DISTRIBUTED BY	

See page 1

S11-00595

NRC027-1695

NRC LETTERS - REVIEW AND APPROVAL REPORT
ED 7159-7

(1) RECORDS MANAGEMENT NO.	(2) SERIAL NO.
	2731 Page 2

(3) SUMMARY (Log No., Title Subject)
Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

(4) COMMITMENT LIST ADDED TO LETTER ☒	(5) PERIODIC / NON-PERIODIC REPORT ☐ YES ☒ NO REPORT NO.
(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC Target Date 8/30/01; Required 9/4/01 ☐ N/A	7) SPECIAL HANDLING ☐ EXPRESS DELIVERY ☒ TELECOPY; DATE SENT
(8) PREPARED BY Rod Cook ext 7782	(9) NOTARY ☒ YES ☐ NO; (10) LICENSE FEE REQUIRED ☐ YES ☒ NO
(11) ADDITIONAL REFERENCES	(12) COMMITMENT NO.(S) CLOSED

(13) COMMENTS

See Page 1

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ DESIGN ENGINEERING MANAGER	D.C. Geisen	DM	8/28/01	8/28/01
☒ PLANT ENGINEERING MANAGER	D.L. Eshelman	See attached		
☒ WORK CONTROL MANAGER	C.D. Nelson	See attached		
☒ DIRECTOR, WORK MANAGEMENT	J. Messina	See attached		
☒ DIRECTOR, TECHNICAL SERVICES	S.P. Moffitt	[illegible] for SPM	8/30/01	8/30/01
☒ DIRECTOR, NUCLEAR SERVICES	L.W. Worley	LW	8/30/01	8/30/01
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DATE ADDED TO LETTER ☐	(15) DATE ADDED BY	(17) ADDITIONAL DISTRIBUTION
DATE SENT TO NRC	(16) DISTRIBUTED BY	
DATE OF BLIND DISTRIBUTION	(18) DISTRIBUTED BY	

See page 1

S11-00596

The NRC Letters - Review and Approval Report (ED 7159-7) should be completed by the Regulatory Affairs Section.

BLOCK 1 RECORDS MANAGEMENT NO. - Regulatory Affairs enters Records Management number prior to distribution of correspondence to NRC.

BLOCK 2 SERIAL NO. - Initiator enters serial number obtained from the Regulatory Affairs Clerk.

BLOCK 3 SUMMARY (Log No., Title Subject) - Initiator enters a summary of the correspondence. This summary should identify if the correspondence is in response to any previous correspondence and why the letter is being written.

BLOCK 4 COMMITMENT LIST ADDED TO LETTER - Preparers checks the block to indiate a commitment list has been included with the letter.

BLOCK 5 PERIODIC/NON-PERIODIC REPORT - Identify whether this correspondence is a Periodic or Non-Periodic Report as identified in Nuclear Group Procedure NG-NS-00807.

BLOCK 6 DATE RESPONSE DUE TO BE SUBMITTED TO NRC - Initiator enters the date the correspondence is due to the NRC. If the correspondence does not have a required due date, the block shall be marked not applicable (NA).

BLOCK 7 SPECIAL HANDLING - Initiator checks if the correspondence requires special distribution to the NRC. If yes, the Regulatory Affairs clerk enters date the correspondence is sent.

BLOCK 8 PREPARED BY - Initiator enters the names of individuals responsible for providing technical information for the correspondence along with his/her name.

BLOCK 9 NOTARY - Initiator checks if the correspondence is required to be notarized.

BLOCK 10 LICENSE FEE REQUIRED - Initiator checks if a license fee is required, per the requirements of 10 Part 170. If yes, the initiator shall complete a Voucher Check Authorization (Form 294) and obtain appropriate fees to accompany the correspondence.

BLOCK 11 ADDITIONAL REFERENCES - Initiator enters any additional NRC correspondence or documents that pertain to the subject correspondence.

BLOCK 12 COMMITMENT NO(S). CLOSED - Initiator enters the Commitment Management System number(s) of any commitments that are closed by the subject correspondence.

BLOCK 13 COMMENTS - Initiator or any reviewer enters appropriate comments regarding the subject correspondence.

BLOCK 14 REVIEW AND APPROVAL - Initiator checks and /or enters the desired reviewer(s) . The technical accuracy of a response to the NRC is the responsibility of the Director and Management individual assigned the action.

BLOCK 15 DATE ADDED BY - Distributor checks the Date Added to Letter Block and signs the Date Added By block to indicate the original letter has been dated prior to distribution to the NRC.

BLOCK 16 DISTRIBUTED BY - Distribution to the NRC shall be made by the Regulatory Affairs Section. Distributor signs the Distributed By block and completes the Date Sent to NRC block.

BLOCK 17 ADDITIONAL DISTRIBUTION - Initiator enters individuals requiring distribution that are not on the standard distribution list.

BLOCK 18 DISTRIBUTED BY - Distributor signs the Distributed By block and completes the Date of Blind Distribution block.

S11-00597

RAS-C152

DOCKETED
USNRC

September 9, 2009 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

FirstEnergy

Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449-9760

Guy G. Campbell
Vice President - Nuclear

419-321-8588
Fax: 419-321-8337

NOT IN ADAMS

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 11
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

Docket Number 50-346

License Number NPF-3

Serial Number 2735

October 17, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC. 20555-0001

Subject: Supplemental Information in Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

Ladies and Gentlemen:

The attached provides supplemental information concerning the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) response (Serial Number 2731, dated September 4, 2001) to Nuclear Regulatory Commission (NRC) Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." Portions of this information were discussed with members of the NRC staff on October 3 and 11, 2001. In addition, the DBNPS and NRC staffs are scheduled to meet and discuss this information and additional NRC crack growth modeling information on October 24, 2001.

This submittal provides updated and additional information in support of the basis for the continued safe operation of the Davis-Besse Nuclear Power Station (DBNPS) until its next scheduled refueling outage commencing in March 2002, at which time the Control Rod Drive Mechanism (CRDM) nozzles and Reactor Pressure Vessel (RPV) head penetrations will undergo qualified visual inspections or appropriate supplemental inspections.

In May 1996, during a refueling outage, the RPV head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed.

DOCKETED
USNRC
September 9, 2009 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

A088

TEMPLATE = SECY-028

DS 03

Accordingly, using the end of the outage in 1996 as the postulated worst-case time for an axial crack to reach a through-wall condition, the projected time for the crack to reach its critical through-wall circumferential size was determined based on the results from an Framatome ANP assessment. This RV Head Nozzle and Weld Safety Assessment demonstrates the postulated crack will take approximately 7.5 years to manifest into an ASME Code allowable crack size. Applying this 7.5 years to the May 1996 inspection projects the worst-case allowable crack size being reached in November 2003. It is important to note the allowable crack size will still maintain an ASME Code safety factor of three.

A Finite Element Gap Analysis was performed by Structural Integrity Associates to verify the gaps between the CRDM nozzles and the RVP head during normal operation would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This analysis concluded that all but four nozzle/penetration interfaces would show visible leakage. These four nozzles are in the least stressed area of the RPV head, and where no leakage attributed to circumferential cracks has been observed at any other plants.

The DBNPS staff is continuing to be involved in and monitoring industry developments regarding CRDM nozzle/penetration cracking, and modifying its inspection plans as appropriate.

Based on the previous inspections conducted, re-reviewed inspection videos, analyses that have been performed concerning crack growth rates, the ability to identify cracking, and industry evaluations and findings, it is concluded there is reasonable assurance that the DBNPS will continue to operate safely to the next refueling outage scheduled for March 2002.

If you have any question or comments, please contact Mr. David H. Lockwood, Manager, Regulatory Affairs, at (419) 321-8450.

Very truly yours,

BWorley

/s

Enclosure and Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, DB-1 NRC/NRR Project Manager
D. Simpkins, DB-1 Acting Senior Resident Inspector
J.A. Zwolinski, NRC/NRR Director, Licensing Project Management
S.S. Bajwa, NRC/NRR Director, Project Directorate III
A.J. Mendiola, NRC/NRR Chief, Projects Section III-2
Utility Radiological Safety Board

RAS C/52

UFFRI

FirstEnergy

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Oak Harbor, Ohio 43449-9760

Guy G. Campbell
Vice President - Nuclear

419-321-8588
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NOT IN ADAMS

Docket Number 50-346

License Number NPF-3

Serial Number 2735

October 17, 2001

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 11
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
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This submittal provides updated and additional information in support of the basis for the continued safe operation of the Davis-Besse Nuclear Power Station (DBNPS) until its next scheduled refueling outage commencing in March 2002, at which time the Control Rod Drive Mechanism (CRDM) nozzles and Reactor Pressure Vessel (RPV) head penetrations will undergo qualified visual inspections or appropriate supplemental inspections.

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A088

DS02

TEMPLATE = SECY 028

Accordingly, using the end of the outage in 1996 as the postulated worst-case time for an axial crack to reach a through-wall condition, the projected time for the crack to reach its critical through-wall circumferential size was determined based on the results from an Framatome ANP assessment. This RV Head Nozzle and Weld Safety Assessment demonstrates the postulated crack will take approximately 7.5 years to manifest into an ASME Code allowable crack size. Applying this 7.5 years to the May 1996 inspection projects the worst-case allowable crack size being reached in November 2003. It is important to note the allowable crack size will still maintain an ASME Code safety factor of three.

A Finite Element Gap Analysis was performed by Structural Integrity Associates to verify the gaps between the CRDM nozzles and the RVP head during normal operation would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This analysis concluded that all but four nozzle/penetration interfaces would show visible leakage. These four nozzles are in the least stressed area of the RPV head, and where no leakage attributed to circumferential cracks has been observed at any other plants.

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If you have any question or comments, please contact Mr. David H. Lockwood, Manager, Regulatory Affairs, at (419) 321-8450.

Very truly yours,

BWorley

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Enclosure and Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
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S.S. Bajwa, NRC/NRR Director, Project Directorate III
A.J. Mendiola, NRC/NRR Chief, Projects Section III-2
Utility Radiological Safety Board

SUPPLEMENTAL INFORMATION

IN RESPONSE TO

NRC BULLETIN 2001-01

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f) and contains supplemental information concerning the response (Serial 2371, dated September 4, 2001) to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for the Davis-Besse Nuclear Power Station, Unit Number 1.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

For: G. G. Campbell, Vice President - Nuclear

By: L W Worley
L. W. Worley, Director - Support Services

Affirmed and subscribed before me this 17th day of October, 2001

Laura A. Jennison
Notary Public, State of Ohio
Laura A. Jennison
My Commission Expires on August 16, 2006.

SUPPLEMENTAL INFORMATION IN RESPONSE TO NRC BULLETIN 2001-01

The Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) submitted its response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" in FirstEnergy Nuclear Operating Company (FENOC) letter Serial Number 2371, dated September 4, 2001. Portions of this information have been discussed with members of the NRC staff on October 3 and 11, 2001.

SUMMARY

This submittal provides updated and additional information in support of the basis for the continued safe operation of the Davis-Besse Nuclear Power Station (DBNPS) until its next scheduled refueling outage commencing in March 2002, at which time the Control Rod Drive Mechanism (CRDM) nozzles and Reactor Pressure Vessel (RPV) head penetrations will undergo qualified visual inspections.

In May 1996, during a refueling outage, the RPV head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed.

Accordingly, using the end of the outage in 1996 as the postulated worst-case time for an axial crack to reach a through-wall condition, the projected time for the crack to reach its critical through-wall circumferential size was determined based on the results from an Framatome ANP assessment. This RV Head Nozzle and Weld Safety Assessment demonstrates the postulated crack will take approximately 7.5 years to manifest into an ASME Code allowable crack size. Applying this 7.5 years to the May 1996 inspection projects the worst-case allowable crack size being reached in November 2003. It is important to note the allowable crack size will still maintain an ASME Code safety factor of three.

A Finite Element Gap Analysis was performed by Structural Integrity Associates to verify the gaps between the CRDM nozzles and the RVP head during normal operation would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This analysis concluded that all but four nozzle/penetration interfaces would show visible leakage. These four nozzles are in the least stressed area of the RPV head, and where no leakage attributed to circumferential cracks has been observed at any other plants.

The DBNPS staff is continuing to be involved in and monitoring industry developments regarding CRDM nozzle/penetration cracking, and modifying its inspection plans as appropriate.

Based on the previous inspections conducted, re-reviewed inspection videos, analyses that have been performed concerning crack growth rates, the ability to identify cracking, and industry evaluations and findings, it is concluded there is reasonable assurance that the DBNPS will continue to operate safely to the next refueling outage scheduled for March 2002.

PLANT DESIGN

The DBNPS has a Babcock & Wilcox (B&W) nuclear steam supply system. The design is similar to other B&W 177-fuel assembly plants, except that DBNPS is of the raised-loop design. The DBNPS has 69 Control Rod Drive Mechanism (CRDM) nozzles of which 61 are used for CRDMs, 7 are spare, and one is used for the Reactor Pressure Vessel (RPV) continuous head vent. Each CRDM nozzle is constructed of Inconel Alloy 600 and is attached to the RPV head by an Inconel Alloy 182 J-groove weld. The DBNPS is unique in the B&W fleet in that it is the only unit that has a RPV head continuous vent that allows for the movement of coolant around the interior of the head, thereby minimizing the stagnation of hot coolant in the top of the head and trapping of air or oxygen.

PREVIOUS INSPECTION RESULTS

In FENOC letter Serial Number 2731, the past inspections of the DBNPS Reactor Pressure Vessel (RPV) head were discussed. As a result of NRC staff questions, supplemental information to and amplification of that discussion is provided in the following.

The inspections performed during the 10th, 11th, and 12th Refueling Outage (10RFO, conducted April 8 to June 2, 1996; 11RFO, conducted April 10, to May 23, 1998; and, 12RFO, conducted April 1 to May 18, 2000) consisted of a whole head visual inspection of the RPV head in accordance with the DBNPS Boric Acid Control Program pursuant to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The visual inspections were conducted by remote camera and included below insulation inspections of the RPV bare head such that the Control Rod Drive Mechanism (CRDM) nozzle penetrations were viewed. During 10RFO, 65 of 69 nozzles were viewed, during 11RFO, 50 of 69 nozzles were viewed, and during 12RFO, 45 of 69 nozzles were viewed. It should be noted that 19 of the obscured nozzles in 12RFO were also those obscured in 11RFO. Following 11RFO, the RPV head was mechanically cleaned in localized areas as limited by the service structure design. Following 12RFO, the RPV head was cleaned with demineralized water to the extent possible to provide a clean head for evaluating future inspection results.

The affected areas of accumulated boric acid crystal deposits were video taped, and have subsequently been reviewed with specific focus on boric acid crystal deposits with reference to the CRDM nozzle penetration leakage as previously observed at the Oconee Nuclear Station, Unit 3 (ONS-3) and at Arkansas Nuclear One, Unit 1 (ANO-1). During the 12RFO inspection, 24 of the 69 nozzles were obscured by boric acid crystal deposits that were clearly attributable to leaking motor tube flanges from the center CRDMs. A further subsequent review of the video tapes has been conducted and corroborates the previous statements and conclusions stated in letter Serial Number 2731 that the results of this review did not identify any boric acid crystal deposits that would have been attributed to leakage from the CRDM nozzle penetrations, but were indicative of CRDM flange leakage. Included as Attachments 2 and 3 are the inspection results for 10RFO, 11RFO and 12RFO, and a figure representing these nozzle locations, respectively. Att 2 3

In summary, results from previous inspections of the CRDM nozzle penetrations provide reasonable assurance for the continued safe operation of the DBNPS until the next refueling outage in March 2002.

ANALYTICAL WORK PERFORMED:

RV Head Nozzle and Weld Safety Assessment

Attachment 4, Framatome ANP's non-proprietary document FRA-ANP 51-5012567-01, "RV Head Nozzle and Weld Safety Assessment," provides an assessment that demonstrates safe operation of the Babcock & Wilcox (B&W)-designed nuclear steam supply systems with the potential for primary water stress corrosion cracking (PWSCC) of RPV head penetration nozzles. The document addresses the assumed presence of PWSCC in either the nozzle base material or the partial penetration welds used in the attachment to the RPV head and the risk assessment with regard to nozzle integrity over a period of time. Att 4

Using the Framatome ANP assessment, the DBNPS feels assured in operating until the next scheduled refueling outage. This is based on the worst case scenario that a visible nozzle axial crack leak developed immediately after start-up from 10RFO in May 1996, and was from one of the 19 drives that could not be inspected in 1998 (11RFO) or the 24 drives that could not be inspected in 2000 (12RFO). The Framatome ANP assessment concluded that such a crack would take 3.5 to 10 years to grow circumferentially through wall. The DBNPS has assumed the 3.5 year value since 3.5 years is based upon multiple crack sites merging together consistent with that which was observed at Oconee 3. This results in the development of a worst case through wall circumferential crack development by November 1999 (May 1996 plus 3.5 years). The Framatome ANP assessment further concluded that this crack would be expected to take an additional 4 years to grow to maximum ASME Code allowable crack size of 270 degrees. Continuing to apply this to the DBNPS's worst case scenario results in the potential to reach a maximum allowable crack size on one of the obscured CRDM nozzles (from 1998 and 2000 inspections) by November 2003. Because this date is beyond the date for the planned March 2002 refueling outage, the DBNPS has concluded that there is reasonable assurance that DBNPS will continue to operate safely to the start of 13RFO, scheduled for March 2002.

Finite Element Gap Analysis

As discussed with the NRC staff during a telephone conference call on October 3, 2001, the DBNPS contracted with Structural Integrity Associates (SIA) to perform a finite element analysis of the RPV head penetrations and nozzles. This analysis was performed to verify that gaps would exist between the CRDM nozzles and the RPV head during normal operation. These gaps would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This plant-specific stress analysis used the DBNPS as-built nozzle and RPV head dimensions. The analysis does not include the effects of primary system pressure in the nozzle gap area that would tend to further open the gaps. The SIA analysis is included herein as Attachment 5 and provides assurance that leakage will be visible on all but four (4) of the sixty-nine (69) nozzle/penetration interfaces. However, the four nozzle/penetration interfaces where it could not be assured that leakage would be visible are nozzle numbers 1, 2, 3, and 4, which are in the center of the RPV head. As documented in the industry history of circumferential cracks observed to date, no leakage attributable to circumferential cracks has been observed in this area from any of the inspections conducted by other licensees. Therefore, based on the verification of inspection results conducted at DBNPS, industry historical results of CDRM nozzle leakage and the finite element analysis performed, it is concluded that no leakage from the CRDM nozzle/head interface has previously occurred at the DBNPS, and through-weld cracking was not present. AHS

INDUSTRY EXPERIENCE & FINDINGS

Since discovery of Alloy 600 cracking at VC Summer and ONS, the DBNPS has been following activities and planning site-specific activities to assure that the Reactor Coolant System pressure boundary integrity is maintained. These activities have included participation in industry groups that are extensively analyzing and characterizing the phenomenological attributes of the cracking issue, and developing sophisticated means of detecting and, as necessary, repairing identified cracks. The findings at other plants are being communicated among the industry in a timely manner which allows aggressive evaluation of the nature, extensiveness and implications of the cracking to ensure the issue is understood as completely as possible, and ensures the development of conservative decision-making. It is through these continuing efforts as well as ongoing plant-specific efforts that the DBNPS can also conclude that there is reasonable assurance that the DBNPS will operate safely to its next refueling outage, scheduled to commence in March 2002.

ALARA ISSUES

In NRC Bulletin 2001-01, page 8, the NRC identified that nozzle penetration activities have the potential for large personnel exposure. Plants have experienced 15 to 40 rem during recent CRDM nozzle activities. The bulletin states that all activities related to the inspection of nozzles should be planned and implemented to keep personnel exposures as low as reasonably achievable (ALARA). As discussed in its initial response to the bulletin, the DBNPS will perform qualified visual inspections or appropriate supplemental inspection of the CRDM nozzle penetrations during its refueling outage scheduled to commence in March 2002. Inspection of these penetrations between now and March 2002, and then again during the refueling outage would significantly increase the personnel exposures. Since the continued safe operation of the DBNPS can be reasonably assured to the beginning of the next refueling outage, completing additional inspections before then would not be consistent with ALARA principles.

CONCLUSION

Based on the previous inspections conducted, analyses that have been performed concerning crack growth rates, the ability to identify cracking, and industry evaluations and findings, it is concluded that there is reasonable assurance that the DBNPS will continue to operate safely to the start of 13RFO, scheduled in March 2002.

Nozzle No.	Core Locat.	Quadrant	1996 Inspection results	1998 Inspection results	2000 Inspection results
			See Note 1.0		
1	H8	1		*Flange Leak Evident*	*Flange Leak Evident*
2	G7	4		*Flange Leak Evident*	*Flange Leak Evident*
3	G9	1		***Flange Leak Evident***	***Flange Leak Evident***
4	K9	2		*Flange Leak Evident*	*Flange Leak Evident*
5	K7	3		Flange Leak Evident	Flange Leak Evident
6	F8	1		**Flange Leak Evident**	**Flange Leak Evident**
7	H10	2		Flange Leak Evident	Flange Leak Evident
8	L8	3		No Leak Observed	No Leak Observed
9	H6	4		No Leak Observed	No Leak Observed
10	F6	4		No Leak Observed	No Leak Observed
11	F10	1		**Flange Leak Evident**	**Flange Leak Evident**
12	L10	2		No Leak Observed	No Leak Observed
13	L6	3		No Leak Recorded	No Leak Observed
14	E7	4		Flange Leak Evident	Flange Leak Evident
15	E9	1		Flange Leak Evident	Flange Leak Evident
16	G11	1		Flange Leak Evident	Flange Leak Evident
17	K11	2		No Leak Observed	No Leak Observed
18	M9	2		No Leak Recorded	No Leak Observed
19	M7	3		No Leak Observed	No Leak Recorded
20	K5	3		No Leak Observed	No Leak Observed
21	G5	4		No Leak Observed	No Leak Observed
22	D8	1		Flange Leak Evident	Flange Leak Evident
23	H12	2		No Leak Observed	No Leak Observed
24	N8	3		No Leak Recorded	No Leak Recorded
25	H4	4		No Leak Recorded	No Leak Observed
26	E5	4		No Leak Recorded	No Leak Observed
27	E11	1		Flange Leak Evident	Flange Leak Evident
28	M11	2		No Leak Recorded	No Leak Observed
29	M5	3		No Leak Recorded	No Leak Observed
30	D6	4		No Leak Observed	No Leak Observed
31	D10	1		**Flange Leak Evident**	**Flange Leak Evident**
32	F12	1		Flange Leak Evident	Flange Leak Evident
33	L12	2		No Leak Recorded	No Leak Observed
34	N10	2		No Leak Recorded	No Leak Observed
35	N6	3		No Leak Recorded	No Leak Recorded
36	L4	3		No Leak Recorded	No Leak Observed
37	F4	4		No Leak Recorded	No Leak Observed
38	C7	4		No Leak Recorded	Flange Leak Evident
39	C9	1		Flange Leak Evident	Flange Leak Evident
40	G13	1		Flange Leak Evident	Flange Leak Evident
41	K13	2		No Leak Recorded	No Leak Observed
42	O9	2		No Leak Recorded	No Leak Recorded
43	O7	3		No Leak Recorded	No Leak Recorded
44	K3	3		No Leak Recorded	No Leak Observed
45	G3	4		No Leak Recorded	No Leak Observed
46	D4	4		No Leak Recorded	No Leak Observed
47	D12	1		Flange Leak Evident	Flange Leak Evident

Nozzle No.	Core Locat.	Quadrant	1996 Inspection results	1998 Inspection results	2000 Inspection results
48	N12	2		No Leak Recorded	No Leak Observed
49	N4	3		No Leak Recorded	No Leak Observed
50	C5	4		No Leak Recorded	No Leak Observed
51	**C11**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
52	E13	1		No Leak Recorded	Flange Leak Evident
53	M13	2		No Leak Recorded	No Leak Observed
54	O11	2		No Leak Recorded	No Leak Observed
55	O5	3		No Leak Recorded	No Leak Recorded
56	M3	3		No Leak Recorded	No Leak Observed
57	E3	4		No Leak Recorded	No Leak Observed
58	B8	1		No Leak Recorded	Flange Leak Evident
59	H14	2		No Leak Recorded	No Leak Observed
60	P8	3		No Leak Recorded	No Leak Recorded
61	H2	4		No Leak Recorded	No Leak Observed
62	B6	4		No Leak Recorded	No Leak Observed
63	B10	1		No Leak Recorded	Flange Leak Evident
64	F14	1		No Leak Recorded	Flange Leak Evident
65	L14	2		No Leak Recorded	No Leak Observed
66	P10	2		No Leak Recorded	No Leak Recorded
67	P6	3		No Leak Recorded	No Leak Recorded
68	L2	3		No Leak Recorded	No Leak Observed
69	F2	4		No Leak Recorded	No Leak Observed

Filed as h/RCS leakage issues/nozzle review Table

Notes:

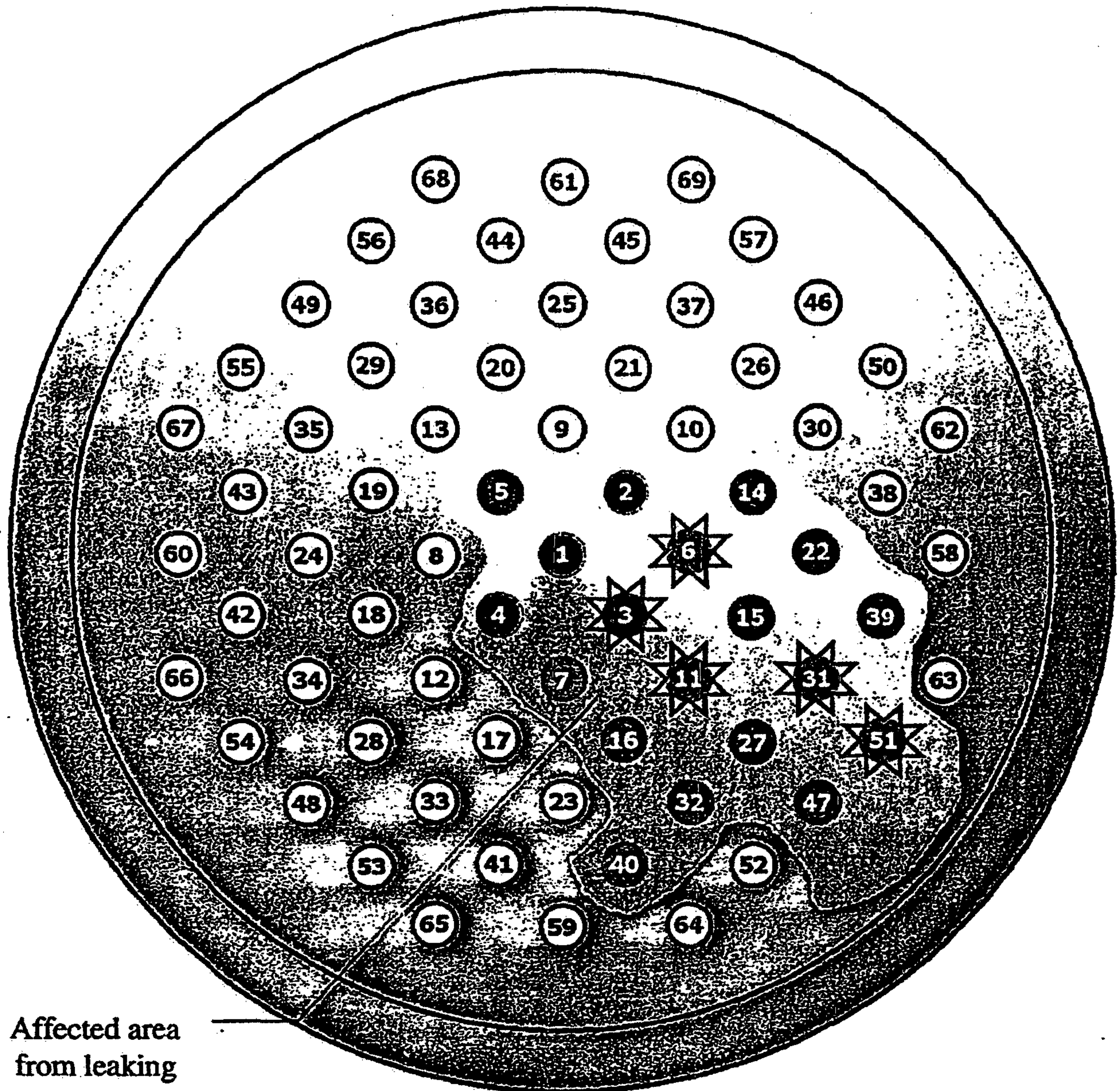
1 In 1996 during 10 RFO, the entire RPV head was inspected.
Since the video was void of head orientation narration, each specific nozzle view could not be correlated.

Bold letters indicate leaking CRDM bolting flanges discovered and repaired during 12 RFO (April 2000).
No Leak Observed = Visual Inspection Satisfactory, No Video Record Required.
No Leak Recorded = Nozzle inspection recorded on videotape
Italicized text indicates nozzles that are not expected to show leakage due to insufficient gap.

RPV Head Inspection Results

3 pages follow

RPV Head 11 RFO Inspection Results



61 - No leakage identified

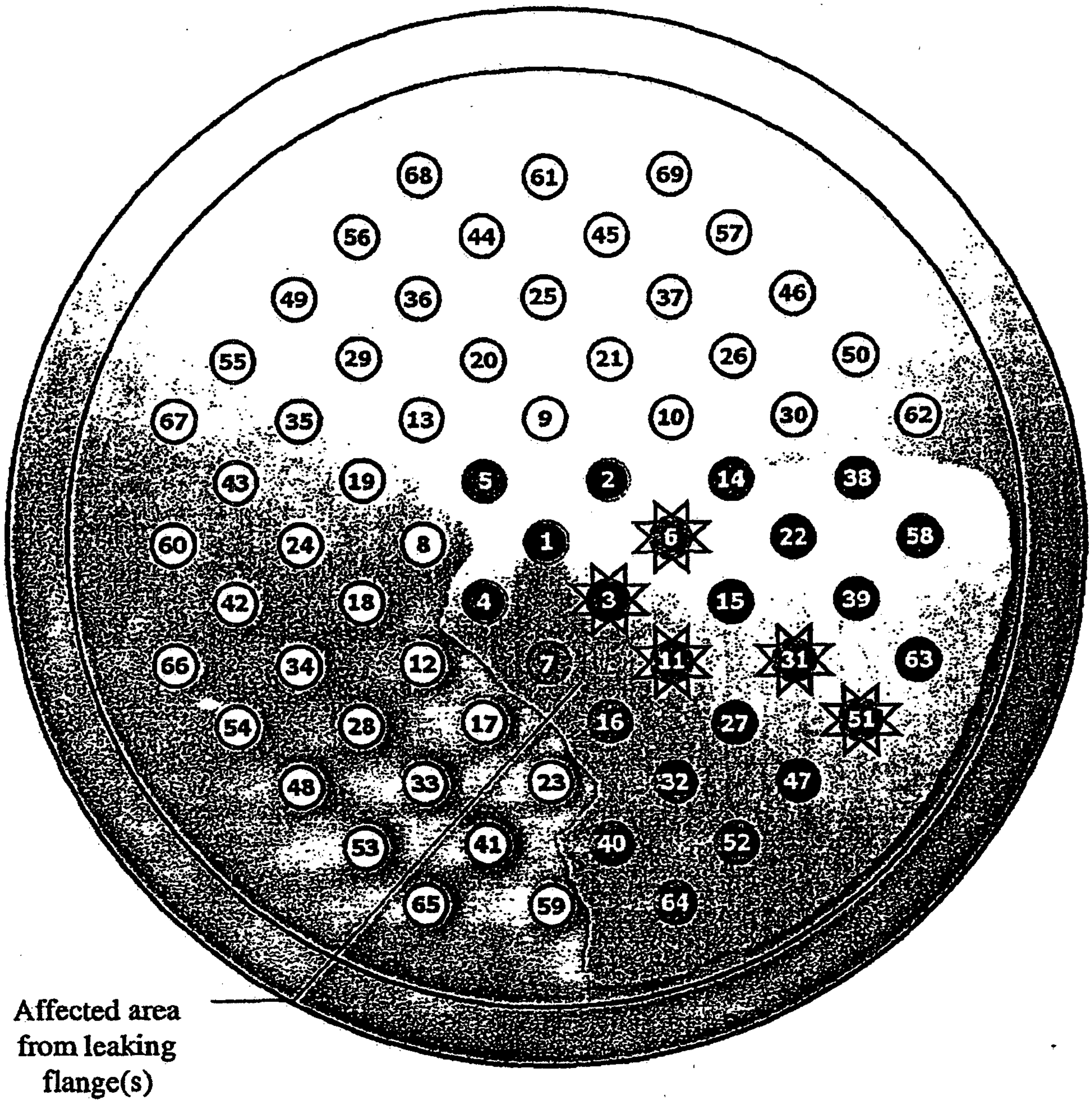
4 - Evaluated not to have sufficient gap to exhibit leakage

3 - Insufficient gap with leaking flange

5 - Nozzle obscured by boron

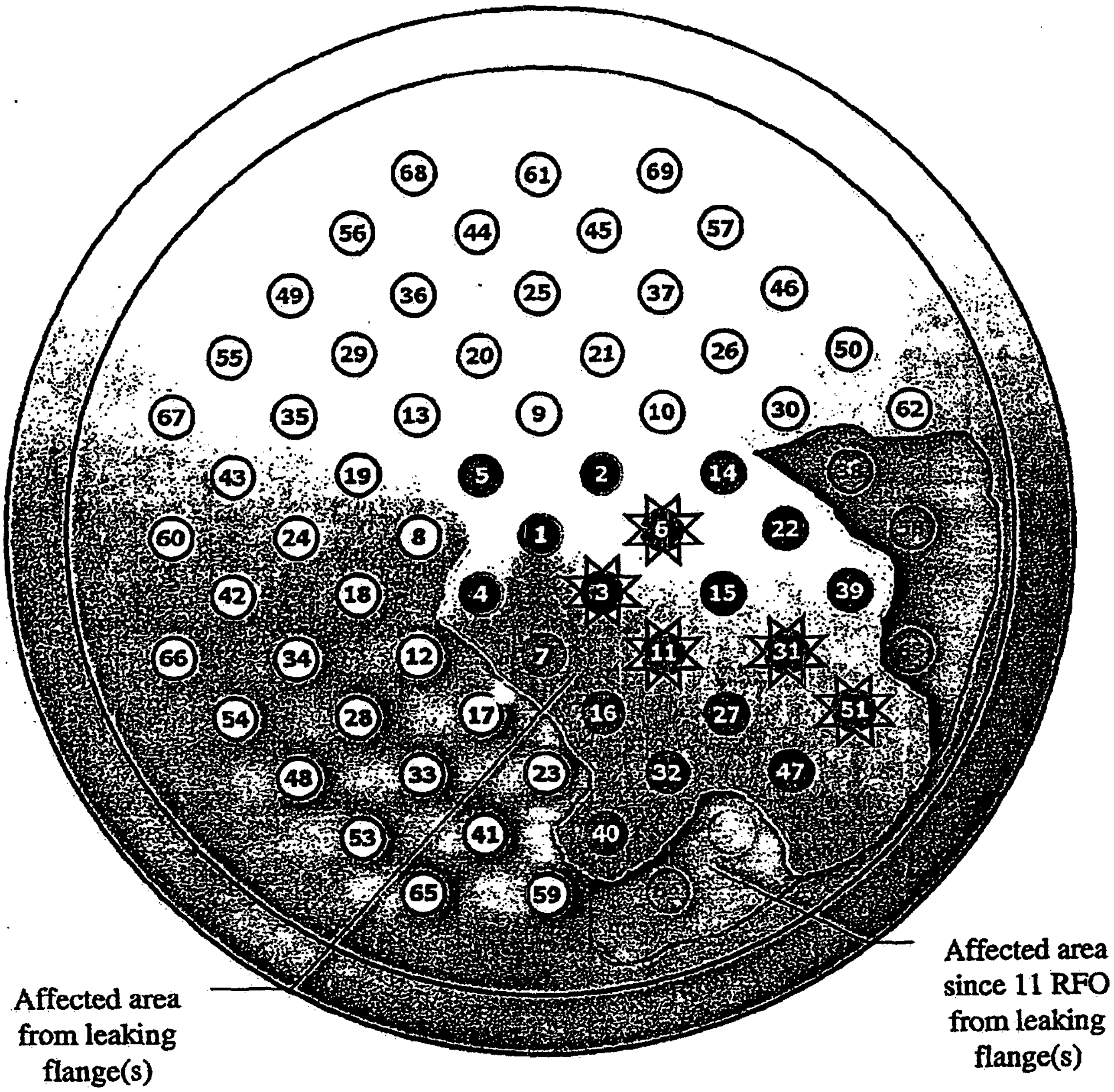
6 - Nozzle obscured by boron with leaking flange

RPV Head 12 RFO Inspection Results



- No leakage identified
- Evaluated not to have sufficient gap to exhibit leakage
- Insufficient gap with leaking flange
- Nozzle obscured by boron
- Nozzle obscured by boron with leaking flange

RPV Head 11 & 12 RFO Inspection Results



- 61 - No leakage identified
- 4 - Evaluated not to have sufficient gap to exhibit leakage
- 3 - Insufficient gap with leaking flange
- 5 - Nozzle obscured by boron
- 6 - Nozzle obscured by boron with leaking flange
- Newly affected, since 11 RFO, by leaking flange(s)

FRA-ANP 51-502567-01, "RV Head Nozzle and Weld Safety Assessment"
Summary Description

The attached document FRA-ANP 51-5012567-01 is a non-proprietary updated version of a previously proprietary FTI document (51-5011603-01). This document is the primary basis document for the DBNPS's assertion that it is acceptable for the plant to continue to operate until its next scheduled refueling outage scheduled to start in March 2002.

The most important portions of this document are Sections 3 and 4.

Fifty-six (56) pages follow

ATT 4
Seval 273?

The attached document FRA-ANP 51-5012567-01 is a non-proprietary updated version of a previously proprietary FTI document (51-5011603-01). This document is the primary basis document for Davis-Besse's assertion that it is acceptable for the plant top continue to operate until it's next scheduled refueling outage scheduled to start in March 2002.

The most important portion of this document is section 4.

It should be noted that Davis-Besse has contracted with SIA and submitted to the NRC SIA's plant specific stress analysis which closely follows the stress analysis performed by FTI for all B&W plants which is covered in section 3.

20440-8 (1/2001)

FRAMATOME ANP

ENGINEERING INFORMATION RECORD

Document Identifier 51 - 5012567 - 01

Title RV HEAD NOZZLE AND WELD SAFETY ASSESSMENT

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This document provides a safety assessment of reactor vessel head nozzles and welds that could potentially be susceptible to PWSCC in B&W-design reactors. Revision 01 incorporates information observed at ONS-2, a risk assessment, and editorial changes.

Table of Contents

1.0 Purpose

The purpose of this report is to provide an assessment that demonstrates safe operation of B&W-design nuclear steam supply systems with the potential for primary water stress corrosion cracking (PWSCC) of reactor vessel (RV) head penetration nozzles. This document addresses the assumed presence of PWSCC in either the nozzle base material or the partial penetration (or "J-groove") welds used in their attachment to the RV head. This safety assessment applies to the RV heads for the following nuclear stations:

Plant[a]	Owner
Davis-Besse (D-B)	First Energy Nuclear Operating Company
Oconee Nuclear Station Units 1, 2, and 3 (ONS-1, -2, and -3)	Duke Energy Corporation
Arkansas Nuclear One Unit 1 (ANO-1)	Entergy Operations, Incorporated
Crystal River Unit 3 (CR-3)	Florida Power Corporation
Three Mile Island Unit 1 (TMI-1)	Exelon Corporation
[a] Note: This group will subsequently be identified as the "B&WOG plants."	

Drawing on the applicable results presented in several B&WOG documents and the results of additional stress, structural, flaw tolerance and fracture mechanics analysis, the objective of this document is met. In addition, the results of a review of the existing safety analyses (Section 8) shows that defense in depth is assured.

2.0 Introduction

Cracking was first observed in a CRDM nozzle at the French pressurized water reactor (PWR) Bugey Unit 3 in 1991. Since that time, the U.S. nuclear industry has developed safety assessments (References 1-3) and several utilities have proactively inspected control rod drive mechanism (CRDM) nozzles considered to be susceptible to PWSCC.

On April 1, 1997, the Nuclear Regulatory Commission (NRC) issued Generic Letter 97-01 (Reference 4). The B&W Owners Group (B&WOG) submitted BAW-2301 (Reference 5) in response to Generic Letter 97-01, which provided details of an integrated inspection plan to address the potential degradation of RV head penetration nozzles at B&WOG plants. [It is noted that the B&WOG plants have two types of RV head penetration nozzles, which consist of CRDM nozzles at all the plants and thermocouple nozzles at ONS-1[a] and TMI-1 only.]

All B&W-design reactors were designed, fabricated, erected, constructed, tested, and continue to be inspected in compliance with 10CFR50.55a (Reference 6). In particular, the RV head penetration nozzles were designed, fabricated, and manufactured to have a low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture in accordance with General Design Criterion 14 of Appendix A to 10CFR50. The Alloy 600 material utilized for these RV head penetration nozzles is an austenitic material that is very ductile and meets the requirements set forth in General Design Criterion 31 of Appendix A to 10CFR50. Finally, accessibility to the RV head is available to assess the structural and leak tight integrity of the RV head penetration nozzles in compliance with General Design Criterion 32 of Appendix A to 10CFR50.

The discovery of the J-groove Alloy 182 weld cracking at ONS-1 and the circumferentially-oriented flaw indications revealed at ONS-3, ONS-2, and ANO-1 have introduced new concerns that must be addressed. This document provides a bounding safety assessment to address the potential severity of these concerns at ONS, ANO-1, and the other B&WOG plants.

2.1 Background

The 1993 B&WOG safety evaluation (Reference 3) presented a stress analysis, crack growth analysis, leakage assessment, and wastage assessment for potential inside surface PWSCC of the B&W-design CRDM nozzles. Based on the results of the stress analysis performed, it was concluded that the peak hoop stresses are greater than axial stresses on the inside surface of the nozzle. Also, the maximum hoop stress is similar for both the center and peripheral nozzles. Thus, if an inside surface crack were to develop in a CRDM nozzle due to PWSCC, the cracks would mainly be axially oriented. It was conservatively concluded that safe operation of the B&W-design plants will not be affected for at least six years (operating with adequate leakage to corrode the RV head), and that within this time, the leak will be detected during a walk-down inspection of

[a] The thermocouple nozzles were removed from ONS-1 at EOC-19.

the RV head area. Thus, the potential for cracking of CRDM nozzles does not present a near-term safety concern.

The same nozzle containing a through-wall crack at Bugey-3 also exhibited an indication of circumferential cracking on its outside surface. In this case, the initiation and propagation of the axial crack preceded exposure of the outer surface of the nozzle above the weld in the annulus to leaking reactor coolant. An addendum to the B&WOG safety evaluation was prepared to address this concern in December 1993 (Reference 7). It was concluded in this evaluation that ample leakage through the penetration would occur to allow detection. In addition, the occurrence of nozzle detachment is highly unlikely during the design life of the B&WOG plants since actions would be taken to repair the nozzle prior to a nuclear safety concern existing.

During a CRDM nozzle inspection at Ringhals Unit 2 in 1992, an indication was detected in the nozzle-to-vessel (J-groove) weld at one penetration. The indication was not indicative of PWSCC; rather, the indication was attributed to a weld defect that occurred during fabrication of the CRDM nozzle to the RV weld. The B&WOG took action to address this concern by acquiring additional data from several sources. First, the data from Ringhals Units 2 and 4 and data from a cancelled Westinghouse reactor, Shearon Harris, were acquired from the Westinghouse Owners Group (WOG). Second, the B&WOG performed an inspection of the RV head from Midland Unit 1, which was a cancelled nuclear station fabricated by B&W.

Another addendum to the B&WOG safety evaluation was prepared to analyze these data (Reference 8). This evaluation included a statistical review and analysis of the J-groove weld inspection data and a stress analysis of the CRDM J-groove weld to determine the minimum weld area that is required to meet the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code primary shear stress limits. It was shown in this report that the maximum areas of weld lack of fusion detected for the Midland Unit 1, Shearon Harris, and Ringhals Unit 2 RV closure heads are well below the ASME Code allowable limits for weld structural integrity. It was concluded that a large margin exists between the statistical bound of the total lack of weld fusion areas in the Midland Unit 1 head and the ASME Code allowable limits. Therefore, the observed lack of fusion areas do not give rise to a safety concern.

In addition, Generic Letter 97-01 requested a description of resin intrusions that may have occurred at the B&WOG plants. The B&WOG response (Reference 5) included a review of plant historical records regarding sulfate excursions. Also, the results of primary water chemistry analysis at each of the B&WOG plants were reviewed for excursions from out-of-specification conditions. Based on these data, it was concluded that the potential for intergranular attack (IGA) or stress corrosion cracking (SCC) of CRDM and thermocouple nozzles was very low.

2.2 B&WOG Plant Inspections

All B&WOG Plants

As described in References 3 and 5, leakage of B&W-design flanges has previously been experienced at each of the B&W-design plants, and visual inspections of the RV head area have been implemented so that flange leaks can be identified and repaired as soon as possible. Primary water that exits from a leaking flange quickly flashes to steam, leaving behind a "snow" of boric acid crystals. Exposure of the RV head to dry boric acid crystals from this type of leakage has not resulted in wastage of the RV head.

The B&WOG utilities have included plans to visually inspect the CRDM nozzle area to determine if leakage is observed on top of the RV head, which would indicate through-wall cracking has occurred, during their outages. In addition, walk-down inspections have been implemented in response to NRC Generic Letter 88-05 (Reference 9) at each of the B&WOG plants. The walk-down inspections include an enhanced visual inspection of the gasket area and RV head during every refueling outage (12-24 months). The B&W closure head and service structure design provides access for a visual or boroscopic examination of the CRDM nozzle area, since the insulation is not resting on the RV head (see Figure 1). If any leaks or boric acid crystal deposits are noted during inspection of the RV head area, an evaluation of the source of the leak and the extent of any wastage is performed. This program has shown to be effective, as evidenced at ONS and ANO-1. These visual examinations provide an acceptable level of quality and safety and are in accordance with 10CFR50.55a and General Design Criteria 30 of Appendix A to 10 CFR50.

BAW-2301 (Reference 5) also describes the ASME B&PV Code Section XI, Article IWB 2500 inspections performed by all the B&WOG plants. In addition, a plant-specific inspection of a CRDM nozzle and a thermocouple nozzle was performed by TMI-1 in 1982 as a result of intergranular attack on the steam generator tubes.

Most recently, NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" (Reference 10) was issued, requesting plant-specific information regarding the structural integrity of the RV head nozzles and extent of leakage and cracking that has been found to date. Information was also requested regarding inspections and repairs that have been completed and those planned in the future to satisfy regulatory requirements, and the basis for concluding that those plans will ensure compliance with the applicable regulatory requirements. Each of the B&WOG member utilities prepared a response that provides this information (References 11-15). A summary of the plant-specific inspections is described below.

Oconee Unit 2

Duke Energy volunteered to perform an inspection[b] of all 69 CRDM nozzles (from the nozzle ID) at ONS-2, which was ranked as one of the B&WOG plants potentially susceptible to PWSCC, in 1994. All indications identified at ONS-2 in 1994 were confined to nozzle number 23 and consisted of 20 indications predominantly axial in orientation. These indications were detected with an eddy current technique and confirmed with dye penetrant testing. An ultrasonic technique could not size the indications on nozzle number 23 because they were too shallow; therefore, the depth was conservatively assumed to be 2 mm (0.079 inch). These indications were subsequently identified as "craze-type" flaw indications. The 1994 eddy current results of a group of eleven other nozzles (numbers 16, 45, 46, 50, 52, 56, 57, 60, 63, and 65) indicated high noise areas, with nozzle number 63 exhibiting the most severe noise of the group. Both ultrasonic and dye penetrant examinations were completed on nozzle number 63 without identifying any indications. Based on this additional information, this group of nozzles with high noise was collectively dispositioned as non-reportable indications.

Both rotating eddy current and dye penetrant examinations were completed on nozzle numbers 23 and 63 during the re-inspection work at ONS-2 in 1996.[c] Multiple indications were observed (i.e., small craze-type flaw indications) that were predominantly axial in nature. Separate eddy current data acquisitions were completed on both nozzles before and after use of a honing cleaning technique to evaluate the effect of the cleaning on the eddy current results. They were confirmed to be in the same location as the high noise areas detected with eddy current in nozzle number 63 in 1994, thus explaining the cause of the noise that was previously unknown.

In 1999, rotating eddy current inspection results for nozzle numbers 23 and 63 at ONS-2 again showed no significant change when the data were compared to the 1994 and 1996 results. Thus it was concluded that the indications had not grown or changed since the 1994 inspection. Rotating eddy current results in 1999 on nozzle numbers 16, 21, 46, 50, 62, and 68 were also obtained and evaluated against the 1994 results. As with all the previous data comparisons, no significant change in the data was observed when compared to the 1994 data.

[b] Non-destructive examination techniques were developed and initially qualified for ID inspection of CRDM nozzles.
[c] Significant development work was completed to improve both the eddy current and liquid penetrant methodologies.

Thus, it was concluded that the indications in these nozzles had not grown or changed since the 1994 inspection.

ONS-2 most recently performed a routine visual inspection of the RV head during a refueling outage in April 2001. Boric acid crystals were observed at four CRDM nozzles (numbers 4, 6, 18, and 30). Liquid penetrant examination identified OD crack-like axial indications below the weld on all four nozzles. Ultrasonic examinations showed that these indications were OD-initiated and that none of the indications were through-wall. An OD-initiated circumferential indication, 0.07 inch in depth and 1.25 inch in length (approximately 36 degrees circumferential extent), was noted above the weld on nozzle number 18 (Reference 11). Eddy current examinations of the ID of the nozzles revealed shallow craze-type flaw clusters in all four nozzles that were distributed around the entire ID circumference (i.e., 360°, above the weld). Based on these results, the leak path was through the interface between the nozzle and the J-groove weld.

The repair at ONS-2 consisted of an automated repair. The four CRDM nozzles were roll-expanded in the upper portion of the RV head in the area of the repair. The bottom portions of the CRDM nozzles were machined out to an elevation above the original structural weld. The machined RV head bores in the area for the new weld and the weld preparations in the CRDM nozzles were liquid penetrant examined to verify that there were no rejectable indications in these areas. The CRDM nozzles were welded to the RV head in accordance with the ASME Code using the temper bead technique using Alloy 52 weld material. The welds were then liquid penetrant and ultrasonic examined. The final operation was to perform an abrasive water jet remediation of the rolled and welded regions.

Oconee Unit 1

As part of a routine visual inspection of the RV head during the ONS-1 refueling outage (November 2000), boric acid crystals were observed at one CRDM nozzle location (number 21) and at five of the eight thermocouple nozzle locations.

Eddy current examination of the inside surfaces of the thermocouple nozzles showed that all eight nozzles contained crack-like indications and that these were predominantly axial in orientation. Ultrasonic examinations from the inside surface of the thermocouple nozzles allowed the weld size to be determined and the axial crack-like indications to be located. Liquid penetrant examination of the J-groove welds (after boring out the nozzles), showed that some cracks had penetrated through the nozzle walls, and that the orientation of these cracks was predominantly axial at the plane where the cracks penetrated into the welds. All eight (8) of these nozzles have been removed by sealing the RV head penetration with a more corrosion resistant material (Reference 11).

Eddy current examination of the inside surfaces of CRDM nozzle 21 and seven other locations (42, 49, 55, 56, 61, 67, and 68) was performed. All eight of the CRDM nozzles contained craze-type indications located in clusters in the uphill region, both above and below the weld. Ultrasonic examinations were performed on the inside surface of 18 nozzles (numbers 17, 21, 22, 28, 34, 42, 47, 48, 49, 52, 54, 55, 56, 61, 62, 66, 67, and 68). No crack-like indications were detected. A liquid penetrant examination was performed on the partial penetration weld of nozzle 21. Two Code acceptable small rounded indications were found. After lightly grinding and performing another penetrant examination, a 0.75 inch radial indication running at a slightly skewed angle across the fillet weld was identified. This crack was ground out of the weld and nozzle material. It extended into the nozzle material approximately 0.4 inch and ran radially out from the nozzle penetrating through the weld and through the butter layer in one location. This crack was identified as the leak source since the annulus was exposed prior to the crack being fully removed.

In summary, cracking was identified in the CRDM nozzle J-groove weld and continued from the weld into the OD of the nozzle. Cracking was also identified in the weld and nozzle ID of the eight thermocouple nozzles. The cracking mechanism was attributed to PWSCC. All indications at these nine (9) locations were removed and weld repairs were performed.

<u>Oconee Unit 3</u>

As part of a routine visual inspection of the RV head during an ONS-3 outage (February 2001), boric acid crystals were observed at nine CRDM nozzle locations (numbers 3, 7, 11, 23, 28, 34, 50, 56, and 63). In addition to observations of through-wall axial flaws above the weld, outside surface circumferential indications (relatively deep and located below the weld) were present on four nozzles (numbers 11, 23, 50, and 56). Also, outside surface circumferential indications above the weld (one through-wall and one nearly through-wall) were present on two nozzles (numbers 50 and 56). Another outside surface circumferential indication above the weld, which was relatively shallow (0.22 inch deep), was present on nozzle number 23. It appeared that these particular cracks initiated from the nozzle OD following exposure to leaking primary water from a through-wall axial flaw (similar to the Bugey-3 cracking). This would require an axial crack, and ultimately a leak path, to either propagate through the weld or the nozzle surface adjacent to the weld. The fact that nozzles 50 and 23 had a circumferential crack at the OD that had not propagated through-wall supports the assertion that an axial flaw is needed prior to a circumferential flaw being formed. The existence of seven (7) other nozzles with at least one axial indication connected to the OD surface of the nozzle at ONS-3 also supports this position.

Liquid penetrant examination of the weld excavation area in nozzle number 34 also revealed a circumferentially oriented flaw indication. Several PT examinations performed during the excavation process revealed that this particular indication spanned about 2-inches in length and was located in the J-groove weld. This indication did not appear to extend to the root of the weld. Shallow axially oriented inside surface indications were also observed in areas of craze-type cracking above and below the weld (similar to those observed at ONS-1 and 2) in virtually all the nozzles examined.

The circumferential cracks discovered at ONS-3 ranged from about 2-mm (0.079 inch) in depth to through-wall. In nozzle number 11, a circumferentially oriented OD crack (below the weld) crossed the path of three axial cracks, the circumferential crack was 0.380 inch deep (61% through-wall), with a 31% circumferential extent (Reference 11). In nozzle number 56, a circumferentially oriented OD crack (above the weld) was through-wall and extended approximately 165° around the nozzle. The circumferential crack in nozzle number 50 was nearly through-wall (i.e., pinhole indications were observed on the ID during liquid penetrant testing) and extended approximately 59° around the nozzle. The circumferential extent and through-thickness depth (from the OD) of nozzle number 23 were 66° around the nozzle and 0.22 inch (Reference 11).

All through-wall cracking observed below the weld appears to have initiated near the toe of the fillet weld. It is most likely associated with the residual weld stresses introduced in this area during manufacturing. Some shallow cracking was also observed on the outer surface at the end of several nozzles (e.g., number 28 and 56).

Ultrasonic examinations were subsequently performed on an additional set of nine nozzles (numbers 4, 8, 10, 14, 19, 22, 47, 64, and 65). From these nine nozzles, eight of them showed no indications. Nozzle number 4, however, showed four shallow axially oriented flaws, all on the inside surface and above the weld. Also, eddy current examinations were performed for these additional nine nozzles. For nozzle number 4, the eddy current results confirmed the findings from the ultrasonic examination. In addition, the eddy current examination revealed shallow craze-type flaw clusters that were found in four nozzles (numbers 8, 10, 14, and 22) and distributed around the entire ID circumference (360°, above the weld).

<u>*Arkansas Nuclear One Unit 1*</u>

Following shutdown for a scheduled refueling outage (March 2001), ANO-1 performed a routine visual inspection of the RV head area. This inspection revealed boric acid crystals in the area of one CRDM nozzle (number 56). Based on the visual inspection results, liquid penetrant (PT), ultrasonic (UT), and eddy current (ET) examinations of the CRDM nozzle and PT of the J-groove weld were also performed. The leak path was determined to be an axial flaw in the nozzle outside diameter that extended beyond each side of the weld. The PT examination identified a circumferential crack, approximately 0.70 inch long in the outside diameter of the CRDM nozzle below the J-groove weld (Reference 12). This crack branched twice and each of the three resulting tributaries extended off-axial (nearly axial) up to the weld fusion line. There is no firm evidence that any cracking occurred in the weld. The UT examination indicated that the subsurface dimensions of the crack extended in a circumferential and then off-axial direction below the weld and in an axial direction through the nozzle past the weld to a termination point 1.3 inches above the weld on the nozzle OD (in the annulus region). The flaw depth dimension was estimated to be a maximum of 0.20 inch into the nozzle wall and thus never penetrating to the nozzle ID surface. It would appear from the NDE evidence that the cracking was confined to the nozzle material, which became the leak path. The ET and PT examination confirmed that the crack had not propagated to the inside diameter of the CRDM nozzle. This flaw is consistent with the PWSCC experience that has occurred at ONS-1. This event also reaffirmed the effectiveness of examining the RV closure head for leaks as means of assuring the avoidance of a safety concern.

3.0 Stress Analysis Efforts

3.1 Summary of Stress Analyses Performed

Nonlinear elastic-plastic finite element analysis was performed in 1993 to characterize stresses in the Alloy 600 CRDM nozzle (SB-167 tube material), the low alloy steel head, the stainless steel cladding in the head, and the Alloy 182 weld material used for the partial penetration weld and butter between the nozzle and head. The purpose of this analysis was to determine the preferential direction for cracking based on the relative magnitude of longitudinal (axial) and circumferential (hoop) stresses. Results were also used to predict crack growth by PWSCC and to support leakage assessments for postulated through-wall cracks in the nozzle wall (Reference 3).

Two bounding nozzle configurations were addressed in the 1993 stress analysis, the center nozzle and one of the outermost peripheral nozzles (hillside nozzle). Taking advantage of full symmetry of the top of the reactor vessel head, the center nozzle was analyzed using a two-dimensional model. Since the outer hillside nozzle penetrates the head at an angle of 38.5 degrees, a 180 degree three-dimensional model was utilized at this location to address the more complicated stress fields associated with an oblique penetration, due in part to ovalization of the nozzle under pressure and thermal loads. The following loading conditions were considered in the 1993 stress analysis to determine long-term sustained stress in the nozzle and weld materials (and to a lesser extent in the head):

1) Shrink fit of the nozzle within the head during installation (0.0010 inch nominal diametric interference).

2) Simulated welding of the nozzle to the head (heatup of the weld material to 2470 °F and cooldown to develop residual stresses).

3) Cold hydrostatic testing of the completed head assembly at a pressure of 3125 psig.

4) Steady state operation at a temperature of 600 °F and a pressure of 2250 psig.

Residual stresses from the welding process are strongly dependent on plastic deformation in the nozzle. Yield strengths for B&W-design plants range from 31 ksi to 64 ksi. For the higher yield strength nozzles, more residual stress is locked in as the weld puddle cools from its molten state. The 1993 stress analysis used the 64 ksi nozzle yield strength as a bounding value.

3.2 Nozzle and Weld Stresses

Since at most locations the inside surface hoop stress is higher than the axial stress, the preferential direction for cracking is axial (in a radial plane relative to the nozzle). Exceptions occur at the lower end of the nozzle and above the weld. Some circumferential cracking may occur on the outside surface of the nozzle, just below the weld, where hoop and axial stresses are similar in magnitude on the uphill side. Axial stresses would also promote the propagation of OD initiated circumferential cracks above the weld.

3.3 Flaw Growth Evaluations

Evaluations of flaw growth from PWSCC have been performed for the J-groove weld and CRDM nozzle as discussed below. Axial ID nozzle flaws were addressed in the original safety evaluation for cracking of B&W-design CRDM nozzles (Reference 3).

3.3.1 Axial J-Groove Weld and OD CRDM Nozzle Flaws

As discussed above, the dominant hoop stress in the J-groove weld would promote axial cracking of this Alloy 182 material. Due to the relatively high crack growth rates observed in autoclave tests with this weld metal in a PWR environment (Reference 16), and considering the increasing stress gradient away from the inside surface of the weld, crack growth through the J-groove weld would be expected. Although the flaw would arrest at the low alloy steel RV head (see Section 4.0), the flaw would continue to grow into the Alloy 600 CRDM nozzle, as seen at ONS-1 (Reference 17).

Calculations were performed to predict the time it would take to grow an axial outside surface flaw (OD flaw) through the nozzle to the inside surface. Assuming a length-to-depth ratio of six, using the Peter Scott crack growth model for Alloy 600 in a PWR environment, and considering the highest stressed location, it would take almost four years for an axial OD flaw that is initially 0.5 mm (0.02 inch) deep to grow through-wall. It has already been reported (Reference 3) that it would take at least four more years for a through-wall flaw to extend two inches above the weld, thereby creating a leak path into the annular region between the nozzle and head.

3.3.2 <u>Circumferential OD CRDM Flaws</u>

Since the OD surface hoop stresses in the weld are about two times the surface axial stresses; flaws originating at this location should be oriented in an axial plane. Development of a leak path through the weld to the annulus between the nozzle and RV head would however, expose the outside surface of the nozzle to the primary water environment. Since there is a band of high axial stress on the outside of the nozzle just above the weld, initiation of a circumferential crack at this location is a concern. Based on experience at ONS-3, the development of an axial leak path through the weld and/or nozzle would precede initiation of a circumferential OD flaw on the outside surface of the nozzle above the weld. Furthermore, as observed at ANO-1, deposits of boric acid crystals on the top of the head would provide evidence of a leak path prior to the initiation of a circumferential OD flaw. For the purpose of performing crack growth calculations, it is conservatively assumed that a small flaw, 0.5 mm (0.02 inch) in depth, initiates immediately after the plant is returned to service. Using 0.5 mm (0.02 inch) as the initial depth of an isolated OD initiated circumferential flaw above the weld, it would take more than 10 years for a short ($l/a = 6$) semi-elliptical surface flaw to grow through-wall. At ONS-3, following the growth of an axial flaw to the annulus between the nozzle and head, there were apparently several initiation sites that linked to form a long circumferential OD outside surface crack above the weld, extending nearly half way around the circumference. Such a flaw could grow through-wall in 3.5 years. Even then, it would take another 4 years for the through-wall flaw to grow another 25% around the circumference. The remaining ligament, which would then be 25% of the original circumference, would still be sufficient to preclude gross net-section failure (nozzle ejection). This ligament satisfies primary stress limits using a safety factor of 3 (Reference 18).

Lack of fusion weld defects between the nozzle and weld, of the type detected at Ringhals Unit 2 and at the cancelled Shearon Harris and Midland plants, should also be considered in light of the potential for CRDM nozzle J-groove weld cracking. This flaw is described as a "wrap-around" circumferential flaw along the cylindrical surface at the nozzle-to-weld interface. As discussed in Reference 8, there may be up to 67% lack of fusion between the nozzle and weld before the ASME Code primary shear stress limits are violated. It has been calculated that it would take two years for a 0.25-inch wrap-around flaw to grow to the 67% limit. This is based on a conservative value of 45 ksi for the average radial stress between the nozzle and weld, and utilizes the high crack growth rates observed in laboratory testing for Alloy 182 weld metal (Reference 16). Based on observations at ONS-1, ONS-2, ONS-3, and ANO-1, where there was no evidence of wrap-around cracking between the nozzle and weld, this is an extremely conservative crack growth prediction.

A 2-inch long circumferentially oriented flaw indication was observed in nozzle 34 at ONS-3. It was located in the weld material and spiraled from a distance of $1^1/_8$ inch from the OD of the nozzle on the uphill side to 0.75 inch from the nozzle as it went about 45° around the weld. Being located in the weld, this laminar-type anomaly is not considered to be a safety concern, since it did not provide a leak path to the environment and it could not lead to ejection of the nozzle.

4.0 Flaw Growth Into the RV Head

A crack, propagating through the J-groove weld by PWSCC, will eventually grow to the RV head (low alloy steel) and the CRDM nozzle (Alloy 600). It is expected that the resultant crack will continue to propagate through the CRDM nozzle material as observed at ONS-1 (Reference 17) and ONS-3, in a direction determined by the residual stress distribution. However, continued flaw growth into the low alloy steel is not expected to occur.

Stress corrosion cracking (SCC) of carbon and low alloy steels is not expected to be a problem under BWR or PWR conditions (Reference 19). SCC of steels containing up to 5% chromium is most frequently observed in caustic and nitrate solutions and in media containing hydrogen sulfide (References 20 and 21). A recent review of literature results was performed by Framatome ANP, which also concluded that SCC of low alloy steel materials is non-credible in PWR environments (Reference 22). Based on this information, SCC is not expected to be a concern for low alloy steel exposed to primary water.

Instead, an interdendritic crack propagating from the J-groove weld area is expected to blunt and cease propagation. This has been shown to be the case for interdendritic SCC of stainless steel cladding cracks in charging pumps (References 23 and 24) and by recent events with PWSCC of Alloy 600 weld materials at ONS-1 and VC Summer (References 25 and 26). Although a PWSCC-initiated flaw may continue to propagate by fatigue crack growth into the low alloy steel head, this is considered to be insignificant over several operating cycles based on anticipated cyclic loads. Since borated water will now be in contact with the low alloy steel, corrosion wastage of the material is expected to occur. This is addressed in Section 6.0 below.

5.0 Leakage Assessment

The B&WOG has performed leakage assessments for various potential leak scenarios expected prior to the recent leak events at ONS-1, ONS-2, ONS-3, and ANO-1. The results from these assessments are documented in detail in Appendix A. The recent experience, however, indicates that the leak rates are apparently very low based on the amount of boric acid crystals observed on leaking nozzles. It was estimated that approximately 0.5 in^3 was present around CRDM nozzle number 21 at ONS-1. In the case of the ONS-1 thermocouple nozzles, five (5) were suspected to have leaks while the other three (3) did not exhibit evidence of boric acid crystals. The examinations subsequently performed on all eight (8) nozzles revealed cracking that would strongly suggest a leak path. It is reasoned that a small leak and narrow annulus can lead to "leak plugging" by the formation of less dense metal oxides in the annulus. Thermal cycling is anticipated to lead to starting or re-initiating a weeping type leak. Therefore, leakage is anticipated to be minimal until a long axial flaw (i.e., approximately the length of the RV head penetration) develops above the weld.

6.0 Wastage Assessment

The purpose of this section is to assess the potential damage that can occur to the RV head as a result of a leaking CRDM nozzle or J-groove weld. Two areas of concern are considered in this discussion:

1) General corrosion damage to the reactor vessel head as a result of exiting boric acid crystals and borated steam condensing on the head insulation from a through-wall crack in a CRDM nozzle or J-groove weld.

2) Corrosion damage both within and in the vicinity of the reactor vessel head penetration due to boric acid corrosion resulting from a through-wall crack in the CRDM nozzle or J-groove weld.

A leaking CRDM nozzle or J-groove weld is of concern because the leaking primary coolant, containing boron in the form of boric acid, can be very corrosive to carbon and low alloy steel materials when subjected to certain environmental conditions. Several studies have been performed to determine these conditions.

A description of the testing performed and their respective results is given in References 3 and 27.

Reference 3 includes a corrosion damage assessment for a variety of conditions and leakage rates assumed to occur with CRDM nozzles. As noted above in Section 5, similar assumptions can be made for the case of leakage that is associated with PWSCC of RV head J-groove welds.

It was determined in Reference 3 that this type of leakage would lead to corrosion of the RV head penetration, at a maximum volumetric metal loss rate of 1.07 in^3/yr. Three defect profiles were postulated to model this level of corrosion for a time period of six years. It was concluded through an ASME B&PV Code evaluation for membrane stresses in the RV head, that safe operation of the plant would not be affected as a result of this level of corrosion of the RV head penetration.

Finally, it was concluded that safe operation of the B&W-design plants will not be affected for at least six years, and that within this time, the leak will be detected during a walk-down inspection of the RV head area. It should be noted that this minimum six-year period represents corrosion of the RV head at the maximum rate of 1.07 in^3/yr, which would only occur when a sufficient leakage rate has been realized. Thus, the potential for cracking of CRDM nozzles and RV head J-groove welds does not present a near-term safety concern. The validity of these assumptions and conclusions was recently verified by the detection of boric acid crystal deposits around CRDM and thermocouple nozzles and the subsequent identification of RV head J-groove weld leakage at ONS-1, ONS-2, ONS-3, and ANO-1. In all cases, only minimal corrosion (wastage) was observed.

7.0 Loose Parts Assessment

As noted earlier, circumferential cracking has been observed on the outside surface of leaking CRDM nozzles at ONS-3. This cracking occurred at the toe of the fillet weld that forms part of the structural attachment to the RV head. In some of these nozzles through-wall axial cracking has also been observed in the nozzle base metal below the weld. Thus, there is a concern that a through-wall circumferential crack could link up with two or more through-wall axial cracks and form a loose part. An assessment of the potential consequences associated with CRDM nozzle fragmentation has been performed (Reference 28). The potential transport of fragments originating at the reactor vessel head penetration were identified and evaluated.

If a piece of the CRDM nozzle were to break away, it could potentially end up in one of three places. The first location is the stainless steel plate around the column weldments (plenum cover) where it would not have an impact on any safety or operational issue in the plant (see Figure 2). The second location is through the gaps around the periphery of the plenum cover and would likely end up in the steam generator, potentially damaging the tubes or tube welds. A fragment lodged within a single tube could, as a result of motion induced by the flow through the tube, cause wear of the tube at the point of contact with the inside surface. Although unlikely, this could eventually result in a small through-wall flaw in the tube, causing a primary-to-secondary leak, which can be detected by monitoring procedures already in place at the plant. Once detected, the plant operators would follow the technical specification action statements to shut down if the leak became significant. This does not introduce any new or unanalyzed event. While this location may cause equipment damage, it is not a safety concern. The third possibility, which could be a safety concern, is that the pieces could enter any one of the 69 column weldments through which the control rod spiders descend (see Figures 5 and 6). It has been calculated that there is a 25% chance or greater for a loose piece to enter one of the column weldments. This is simply based on an area ratio of the column weldments in the upper head and the fact that low cross flow velocities in this region would tend to allow debris to fall vertically. In addition, the leadscrews could tend to guide the debris such that the probability of entering the column weldment may be much higher than 25%. If fragments enter the column weldments, they will likely be stopped on one of the control rod guide tube brazements where relatively small fragments (< ¾ inch) would be capable of precluding complete control rod insertion.

Based on experience at ONS-3, circumferential and axial cracking below the weld is accompanied by through-wall axial cracking at and above the weld. The ONS experience coupled with the extensive examinations performed in Europe, and the stress analysis results described in Section 3.0 indicate that the predominant cracking orientation is axial.

In addition, there have been a total of 27 non-leaking nozzles at both ONS-1 and ONS-3 subjected to both eddy current and ultrasonic examinations. Very shallow craze-type cracks were revealed above and below the welds. No OD cracks were detected at the nozzle-to-weld intersection (below the weld) for these 27 nozzles. In each case, these nozzles were found to be free of cracking. These observations and results support the assertion that there is a high probability that detectable leakage would precede the development of a loose part.

8.0 Safety Analysis Review

In this section, the plant safety analyses will be reviewed to determine if a safety issue exists and to provide justification that the consequences of a failure of a single CRDM nozzle are bounded by the existing plant safety analyses and will support plant restart and continued operation.

Loss of coolant accidents (LOCA) and non-LOCA safety analyses are performed to justify that the nuclear power plants can be safely shut down following postulated accidents. Although these analyses do not specifically consider failure (i.e., complete severence) of a CRDM nozzle, they consider events that have more limiting consequences. LOCA analyses typically postulate breaks in RCS pipes from those within the plant makeup capacity up to and including a double-ended guillotine break of the hot leg to demonstrate acceptable core cooling in the short term as well as the long term. Non-LOCA safety analyses specifically postulate a control rod ejection accident, although the CRDM nozzle remains intact. The rod ejection event postulates that the CRDM flange bolts fail and the control rod is ejected out of the CRDM housing. These plant safety analyses are reviewed in the following paragraphs to determine if a more substantial safety issue exists based on the leaks that have been observed at ONS-1, ONS-2, ONS-3, and ANO-1. Where applicable, additional margin is identified to further support plant restart and continued safe operation.

As described in the previous sections, once a crack initiates, it is estimated that it may take up to six years for it to migrate through the CRDM components and begin to leak at undetectable rates. Detection of such minor leaks that grow at slow rates is by visual inspections of the CRDM nozzles as noted with the ONS-1, ONS-2, ONS-3, and ANO-1 outages. These routine inspections of the potentially affected areas will identify if any leak has initiated well before the weld or component could fail catastrophically. The detected cracks have grown predominantly in the axial direction, although some circumferential cracks have been observed near the weld. These as-found circumferential and axial cracks have been evaluated, and it was concluded that the structural integrity of the component retains sufficient margin to ensure continued safe operation of the plant. In addition, the maximum projected growth rate from the boric acid corrosion of the RV head penetration from a minor leak would not propagate into adjacent CRDM nozzle failures. Therefore, simultaneous catastrophic failure of multiple nozzles will not be postulated.

Since failure of multiple CRDM nozzles is not considered credible, the primary concern is the failure of a single nozzle. This unlikely, yet postulated failure leads to RCS inventory loss and less core shutdown margin for the plant safety analyses. These aspects are addressed in the following paragraphs relative to the consequences already included in the existing LOCA and non-LOCA plant safety analyses.

Loss-of-Coolant Accident

Plant LOCA analyses do not specifically analyze the potential failure of the reactor vessel or any of the attached nozzles, but they do postulate break sizes from 0.01 ft^2 to 14.2 ft^2 in area in any RCS pipe. A break in a CRDM from a crack that formed, propagated without detection, and failed catastrophically would be bounded by the RCS inventory losses considered in the existing plant LOCA analyses. Also, this break location is favorable from a core cooling standpoint, in that it is on the hot side of the core, such that no emergency core cooling system (ECCS) fluid is bypassed directly out of the break. That means that all the ECCS liquid is available for core cooling. The core shutdown for this event is assured by the insertion of the remaining control rods, augmented by the soluble boron reactivity control via the boron in the ECCS injection fluid.

Despite the fact that the existing LOCA analyses bound the CRDM nozzle failure with respect to inventory loss, there remains additional margin based on the credited rod worth and the RCS leakage detection systems. In the small break LOCA analyses, minimum control rod worths are credited. The control rod of highest worth is assumed to be stuck out of the core, and only a fraction of the remaining worth is used in demonstrating that at least a 1 percent shutdown margin exists at hot zero power conditions.

The RCS leakage detection systems are required by the plant technical specifications to detect unidentified leak rates of 1 gpm or greater. If the leak rate is higher, the plant will be shut down, and a controlled cooldown will be initiated. The makeup system will provide sufficient inventory and boron control. Insertion of the control rods will not be inhibited, and the core reactivity will be controlled. Following reactor shutdown, the consequences of a CRDM nozzle failure are decreased, thereby providing additional assurance that a safe shutdown is not compromised by the leakage that has been found or postulated to propagate during a single operating cycle with a leak in a CRDM nozzle.

Non-LOCA Safety Analyses

The plant non-LOCA safety analyses, for which consequences can be more severe if the core is not completely shut down, assume that the highest worth control rod is stuck out of the core, and at least a 1 percent shutdown margin exists at hot zero power conditions. Also, the consequences of a control rod ejection accident (CREA) are explicitly analyzed and included in the individual plant Final Safety Analysis Report (FSAR). Limitations are also imposed on each core design to limit the worth of any ejected control rod worth at hot full power to a value much less than the value assumed in the accident analyses.

The standard NRC-approved methodology (for Framatome ANP) consists of (1) calculating the maximum single ejected rod worth throughout cycle life, (2) verifying that the limits bound these maximum worths after augmenting by a 15 percent uncertainty, and (3) verifying that the core operating (rod index) limits preserve the calculational basis of the maximum worth. Because the typical analysis methodology uses the core average power response, the results of the calculation are sensitive to the total amount of reactivity inserted, not the number of control rods ejected. Consequently, the existing analysis will remain bounding for any number of ejected control rods, provided the total reactivity inserted into the core remains less than the values analyzed and reported in the FSAR. This provides additional margin, such that the consequences for the unlikely failure of a single CRDM nozzle will not be more severe than that already considered by each new fuel cycle for a limiting control rod ejection accident scenario.

9.0 Risk Assessment for CRDM Nozzle Cracks

The purpose of this section is to provide a risk analysis to supplement and support the deterministic safety assessment. The other sections of this safety assessment report describe the traditional engineering assessment of the CRDM nozzle cracks, including deterministic issues such as the impact upon safety margins and defense-in-depth. This deterministic analysis provides the source material upon which the risk assessment is based. This risk analysis estimates the core damage frequency (CDF) associated with operation with potentially undetected CRDM nozzle cracks, such as those found recently at ONS.

9.1 Potential Risks from CRDM Nozzle Cracking

Potential risks associated with the possibility of undiscovered CRDM nozzle cracks include:

- LOCA due to CRDM nozzle rupture or detachment
- Anticipated Transient Without Scram (ATWS) due to rod ejection accident (one or multiple)
- ATWS due to loose parts blocking control rods
- Damage due to CRDM and nozzle missile during accident

Of these, LOCA is considered to be the most important from a core damage frequency or risk perspective.

The random nature of crack initiation and growth makes it highly unlikely that multiple circumferential cracks will reach critical size in different CRDM nozzles at the same time. This assertion is made because the mean-time-to-failure in any given CRDM nozzle population is randomly (and widely) distributed. Even if there were several CRDM nozzles with unrevealed degradation, the loads administered during a plant transient would not impact them in identical ways. Because of nonhomogeneous crack initiation and growth, one CRDM nozzle failure time would precede the other(s). The recent B&WOG plant experience (see Sections 2.2 and 9.2) supports this assertion. The evidence of crack extent for the observed OD circumferential cracks above the J-groove weld indicates a random distribution of crack lengths. Therefore, simultaneous crack-initiated failures of redundant CRDM nozzles are very unlikely, and the risk from multiple CRDM nozzle failures due to cracking is very small.

In addition, the plant shutdown margins (if evaluated realistically) are such that several CRDM nozzle failures could be tolerated before the risk would increase over that of a single CRDM nozzle failure. Even with conservative success criteria for reactor trip, two or three CRDM nozzle failures could easily be tolerated from a reactivity standpoint. Therefore, it is concluded that the reactivity accidents (rod ejection or control rod blockage by loose parts) are not credible risk contributors, because of the number of simultaneous CRDM nozzle failures that would be required.

Missiles generated by CRDM nozzle failures are also not credible risk contributors. Even in the unlikely event of a CRDM nozzle detachment, the missile shields will prevent consequential damage to the reactor building or other safety systems.

Therefore, the risk impact that will be addressed and quantified is the risk from a LOCA. This analysis will estimate the incremental CDF due to a LOCA caused by a CRDM nozzle that may fail during operation due to undiscovered cracks.

9.2 Identification of CRDM Nozzle Cracks that are a Risk Concern

Of particular concern are circumferential cracks above the weld. A circumferential crack of sufficient extent may cause a large leak or a LOCA due to gross structural failure (net-section collapse) of the CRDM nozzle pressure boundary. The OD of the CRDM nozzle just above the J-groove weld (which is normally dry) is the only region on the CRDM nozzle pressure boundary where there is high axial stress relative to hoop stress. This region is susceptible to circumferential PWSCC cracks only if there is a source of primary water to the nozzle penetration annulus, such as might occur if there is a through-wall (TW) axial crack initiated from the ID of the CRDM nozzle or a crack in the J-groove weld.

Although axial CRDM nozzle cracks and J-groove weld cracks have occurred, they are not likely to result in a significant LOCA directly. The primary risk concern with the axial cracks and weld cracks is that the primary water leakage through the crack can provide moisture to the CRDM nozzle exterior and promote OD PWSCC.

Recent B&WOG experience (see Section 2.2) has included several axial cracks propagating through the J-groove weld or the area of the CRDM nozzle near the weld. At ONS- 3, there were nine CRDM nozzles with TW axial cracks above the J-groove welds or in the welds themselves. These cracks provided a path for primary water to the OD of the CRDM nozzle (in the annulus area just above the weld) where subsequent inspection indicated that three of these had indications of OD circumferential cracks above the weld. At ANO-1, there was an axial OD crack in the nozzle below the weld (i.e., in the area that is normally wetted) that extended to above the weld on the nozzle OD, thus wetting the OD area above the weld. At ONS-1, there was a crack through the J-groove weld of a CRDM nozzle that wetted the OD of the nozzle in the annular region above the weld. And at ONS-2, four CRDM nozzles were found with axial cracks that caused leakage to the annular region. Of these, one had indications of an OD circumferential crack above the weld. This experience suggests that there is a risk of a LOCA-sized CRDM nozzle failure from OD circumferential cracks that may be initiated due to primary water leaking into the annulus area from undetected ID-initiated TW axial, other OD-initiated axial, or J-groove weld cracks.

The four above-the-weld OD circumferential cracks at ONS-2 and ONS-3 were repaired along with the other crack indications. Excavations to clear these indications extended up to 180 degrees in the circumferential direction, and complete characterization of the indications was not recorded due to the aggressive nature of the excavations. However, subsequent examinations of ultrasonic test (UT) data taken before the excavations have indicated the circumferential extents of the cracks to be approximately 36 degrees (ONS-2 nozzle 18), 66 degrees (ONS-3 nozzle 23), 59 degrees (ONS-3 nozzle 50), and 165 degrees (ONS-3 nozzle 56) (see Section 2.2).

It is also possible that an ID-initiated crack could grow circumferentially to fail the CRDM nozzle pressure boundary directly. These cracks have been considered in this risk assessment. However, the operating history and probabilistic fracture mechanics analysis of ID-initiated circumferential cracks indicates that the likelihood of this failure mode is very small due to the nature of the stresses on the ID of the CRDM nozzle. To support this assertion, a probabilistic fracture mechanics prediction was made (Reference 29) of the ID-initiated TW crack frequency using the CHECWORKS computer code (Reference 30), the EPRI tool for predicting time to Alloy 600 PWSCC. The CHECWORKS analysis shows that the expected frequency of ID-initiated circumferential cracks is much less than the expected frequency of ID-initiated axial cracks. For the worst-case B&WOG plant, CHECWORKS predicts a median cumulative probability of 0.07 over the (60 year) plant life of getting an ID-initiated above-the-weld TW circumferential crack. This is a frequency of approximately 0.002 per reactor-year if averaged over the remaining plant life. That frequency is insignificant relative to the probability of axial cracks that may contribute to OD PWSCC (which is discussed further in Section 9.3.1). Therefore, the focus of the risk assessment is on the scenarios for circumferential OD CRDM nozzle cracking, and the risk estimated for OD-initiated circumferential cracks is considered representative of the overall risk.

With respect to impact upon risk, the only CRDM nozzle cracks that are risk significant are those where detectable symptoms of the degradation are not identified (and acted upon) prior to total failure of the nozzle. The risk analysis discussed below estimates the probability that CRDM nozzle failure will occur before a successful visual inspection detects telltale boron crystals on the exterior of the reactor vessel head.

9.3 OD Circumferential Crack Risk

The risk analysis focuses on scenarios in which an OD circumferential crack can grow to failure, causing a LOCA. The OD circumferential crack grows on the CRDM nozzle pressure boundary as a result of PWSCC caused by CRDM nozzle of J-groove weld leakage that wets the exterior of the CRDM nozzle in the annulus around the head penetration. The incremental core damage frequency for a LOCA induced by OD circumferential CRDM nozzle cracking is the product of the following factors:

- Frequency of weld or nozzle leak that wets OD of CRDM nozzle in the susceptible location
- Probability that CRDM nozzle leakage is undetected
- Time-dependent probability that total failure of CRDM nozzle will occur due to undetected crack initiation and growth on nozzle OD
- Probability of core damage from resulting LOCA

These events are shown as headers on the event tree (Figure 5) in which sequences that result in core damage are shown. Estimates of the event tree probabilities and initiating event frequency are provided in the following sections.

9.3.1 Probability of Weld or Nozzle Leak

In this section, the frequency of CRDM nozzle leaks that may wet the OD above the weld is estimated. It is assumed that some CRDM nozzles may be in service with near-TW nozzle cracks or weld cracks that may surface in the next fuel cycle. The CRDM nozzle cracks of interest are those above weld axial cracks and weld cracks that may leak primary water to the exterior of the CRDM nozzle in the annulus region of the RV head penetration.

The CRDM nozzle leak rate has been estimated from the recent inspection experience at ONS-1, ONS-2, ONS-3, and ANO-1. At these four plants, boron crystal deposits indicated leakage at 15 CRDM nozzles, including one at ONS-1, four at ONS-2, nine at ONS-3, and one at ANO-1 (see Section 2.2). It is uncertain how long these CRDM nozzles have been leaking. For the purpose of estimating a leak frequency, it will be assumed that half of these 15 leaks appeared during the most recent fuel cycle and half in the previous fuel cycle. It is likely that some of these leaks were actually present in earlier refueling outages, but were not identified as nozzle leaks at that time. Therefore, 15 leaking CRDM nozzles in approximately twelve plant-years (four plants times two cycles times 1.5 years per fuel cycle), gives an average frequency of approximately 1.25 leaking CRDM nozzles per reactor-year.

A prediction was also made (Reference 29) of the ID-initiated TW crack frequency using CHECWORKS (Reference 30). CHECWORKS is an empirical code, and the recent inspection results were taken into account by adjusting the crack initiation reference times. The results of the CHECWORKS analysis of ID-initiated cracking are dominated by the contribution from axial cracking, which was over two orders of magnitude more likely than circumferential ID cracking to cause a TW crack above the weld. The results of this analysis predict a median frequency of ID-initiated TW cracks above the J-groove weld of 0.52 per reactor-year averaged over the remaining plant life (assuming 60 year life). However, the CHECWORKS analysis is for ID-initiatec nozzle cracking only. CHECWORKS (as it is now configured) is not designed for OD-initiated cracks above the weld or for J-groove weld cracking. Hence, as a prediction for CRDM nozzle leak frequency, it may underestimate. Therefore, to be conservative the mean value of 1.25 CRDM nozzle leaks per reactor-year, which was estimated from the plant experience, will be used in the risk assessment.

9.3.2 Probability that CRDM Nozzle Leakage is Undetected

A human reliability analysis (Reference 31) has been performed to estimate the human error probability (HEP) for the utility's inspection personnel failing to detect boron crystal deposits on the RV head that are indicative of a CRDM nozzle leak. CRDM nozzle leakage will be detectable through the accumulation of boron crystals on the top of the RV head around the base of the affected CRDM nozzles. It is assumed that any CRDM nozzle crack is undetectable until a through-wall crack (or weld crack) deposits boron crystals on the exterior of the RV head.

For OD-initiated above-the-weld cracks, the fracture mechanics model that is most relevant to the human reliability analysis is how long it takes, once wetted, for an OD crack to initiate and grow to the critical size for CRDM nozzle failure. This time (estimated in Section 9.3.4) will indicate how many opportunities (refueling outages) there may be to detect the boron crystals before total failure of the CRDM nozzle. Another factor important to the risk assessment is when the boron crystal deposition will be visible relative to the growth of the circumferential cracking.

The OD PWSCC failure mechanism requires a moist environment from the presence of primary water in either the liquid or steam state. Primary water from a leaking CRDM nozzle or weld will be deposited into the nozzle penetration annulus. During steady state operation there is a small radial clearance in the annulus above the weld to the surface of the RV head (see Appendix A). The primary leakage into the annulus may initially be very small, as might be the case for a pinhole leak, or somewhat larger, but there can only be PWSCC when there is a sufficient rate of leakage to keep the annulus area moist. Very small leaks are not likely to provide the appropriate environment in the annulus initially, considering the temperatures and pressures on top of the reactor vessel. The rate of boron crystal deposition will also be dependent upon the size of the leak. Moderate-sized leaks will generate boron crystals rapidly. For smaller leaks, there may be some time before significant boron crystals accumulate. However, as the boron crystals build up in and around the annulus, their presence will tend to trap moisture below. It is also possible that for a small leak there may be intermittent "leak plugging" and a weeping type leak (see Section 5) as the buildup of boron crystals intermittently "vents." Hence, it is reasonable to conclude that the environment required for the initiation of PWSCC on the OD of the CRDM nozzle (i.e., above the weld), whether it be from a small or moderate leak, will coincide roughly with the presence of visible boron crystal deposits.

9.3.2.1 Reactor Vessel Head Inspections

As a result of Generic Letter 97-01 (Reference 4), the B&WOG licensees have made a commitment to perform timely inspections of CRDM nozzles (and other vessel closure head penetrations). This commitment is maintained by permanent addition of a task item/work order into the refueling outage schedule program. Discovery of (new) boron on the head will result in the finding being placed in the licensee's Corrective Action Program (CAP).

CRDM nozzle flange leaks have occurred on several past occasions at B&WOG plants. Boric acid crystal buildup from these leaks may have masked indications of CRDM nozzle leakage in the past, and may have contributed to the exterior circumferential OD cracks at ONS not being detected by an inspection sooner.

CRDM nozzle flange leakage in the past was not considered to be unusual; however, once discovered, the CRDM nozzle flange was repaired to stop the leakage. Part of the repair process was to replace the gasket. The B&WOG licensees have been gradually repairing flanges and replacing gaskets since about May 1989. To date, nearly all of the B&WOG plant CRDM nozzle flange gaskets have been replaced with a stainless steel/graphite gasket, which, according to operating experience, are less prone to leakage. The number of CRDM nozzle flanges that still have the old gaskets is currently quite small (total of about a dozen over all of the B&WOG plants). Any flange leakage from one of these few remaining old-style gaskets would be quite evident, and would be promptly addressed.

Over the last five to seven years, the RV head inspections have become increasingly more meaningful because of utility efforts to clean the head of boron deposits resulting from past CRDM nozzle flange leakage and other sources. A clean RV head will make new boron crystals at the nozzle penetrations more evident, and reduce the likelihood that the leakage will be missed or masked by other sources of boron on the RV head.

The method of RV head inspection for indications of boron varies among the B&WOG plants. The methods vary from a simple visual inspection to the use of a mobile RV head robot with an attached video camera. However, none of the B&WOG plants have insulation directly on the reactor vessel head that may impact visual inspections. With all of the methods, the RV head inspection process is simple and straightforward, such that a written procedure is not necessary for a successful inspection. For the visual method, the RV head is observed through eight or nine access panels in the service structure with a high intensity portable light. The farthest an inspector would be from a CRDM nozzle is five feet. To ensure completeness, the inspection is carried out with a paper map of CRDM nozzle locations. The visual inspection method requires approximately two hours to complete. Other methods, such as use of a boroscope (i.e., camera on a stick) or RV head robot, result in a permanent record of the inspection on videotape. These methods also rely on the use of a paper map of CRDM nozzle locations to ensure completeness of the inspection.

9.3.2.2 <u>Estimate of Human Error Probability for Visual Inspections</u>

HEPs have been estimated for failure of the visual inspections using a combination of the Human Cognitive Reliability Model (Reference 32) and Swain and Guttman's Handbook (Reference 33). Since, visual inspections will occur with each refueling outage, a time-dependent failure probability is estimated considering the inspections to occur at two-year intervals. This is conservative since refueling cycles range between 18 and 24 months.

The human reliability analysis considered three ways in which the inspection process can fail to detect the boron crystals that are indicative of a CRDM nozzle leak. These include failure to conduct the inspection, failure to observe the boron crystals on the RV head when present, or failure to identify boron crystals resulting from a CRDM nozzle leak due to masking by other sources of boron (i.e., from CRDM nozzle flange leakage).

The human reliability analysis estimates that the HEP for failure of the visual inspection to detect signs of CRDM nozzle leakage at the first opportunity is 6.0×10^{-2}. The human reliability analysis also estimates the failure probability for a second and third inspection (spaced at refueling outage intervals) of finding the same leaking CRDM nozzle, given failure of the previous inspection(s). The probability of repeatedly failing to detect the boron deposits at consecutive inspections (assumed two-year intervals) is a dependent relationship. The impact of this dependency (aside from the crack growing larger) is that the boron deposit will be more prominent and more difficult to miss at the next inspection. However, there is also the possibility that errors made in previous inspections will be repeated, the most important error being failure to perform the inspection. The human reliability analysis balances these competing dependencies, and appropriately adjusts the HEP with each subsequent outage. After failure of the first visual inspection, the dependency for repeatedly failing to perform the inspection is conservatively assumed to be stronger than the dependency of the boron deposit being more evident with time, thus causing the HEP to increase with additional opportunities. The human reliability analysis estimates that the HEP for failure to detect the CRDM nozzle leakage on the second opportunity is 6.5×10^{-2} and that it is 0.11 for the third and each subsequent outage. These HEPs are conservative considering the increased future emphasis on effective visual inspections of the reactor vessel head penetrations. Conservative HEPs have been used to encompass the uncertainty that is generally present in HEP estimates.

9.3.3 Probability of OD Crack Initiation

The time-to-OD-crack-initiation, once the exterior of the nozzle is wetted with primary water, is unknown. Computer codes used to predict time-to-PWSCC initiation are unreliable for OD PWSCC because the environment on the exterior of the CRDM nozzle (i.e., exterior to the pressure boundary) may be different than on the ID of the nozzle, especially in terms of boron concentration and length of time wetted. Therefore, to be conservative, the risk analysis assumes that the time-to-OD-crack-initiation is zero for all CRDM nozzles with exterior primary water wetting.

This approach is conservative with respect to the observations of the 15 leaking CRDM nozzles that were recently found at ONS and ANO-1. Only four of these

CRDM nozzles had indications of OD circumferential cracking above the weld. It is unknown specifically how long each of these CRDM nozzles has been leaking or whether OD cracks would have initiated on the others if leakage had continued undetected. Therefore, since a valid time-dependent model for OD PWSCC crack initiation is unavailable, it is conservative to assume that OD crack initiation will occur in 100% of the CRDM nozzles that have leakage into the annular region above the weld.

This approach (100% crack initiation with zero time-to-initiation) bounds the uncertainty associated with the lack of probabilistic fracture mechanics data for OD PWSCC crack initiation.

9.3.4 <u>Time to Total Failure of CRDM Nozzle</u>

A probabilistic fracture mechanics analysis (Reference 34) was performed to determine the probability of net-section failure of CRDM nozzles after initiation of above-the-weld OD circumferential cracking. The probabilistic fracture mechanics model was built around the deterministic crack growth model (Reference 18) described in Section 3.3.2. The crack growth model uses the Peter Scott model with worst case stresses. The probability of gross net-section failure is determined by performing a Monte Carlo simulation on a typical B&W-designed CRDM nozzle by varying the defining parameters of crack growth and size used in the deterministic fracture mechanics analysis. Available industry data were used to define distributions for key variables and conservatism was used where the data were sparse.

For the initial flaw distribution, the calculation was performed using parameters representative of the nozzle cracks found at ONS. For example, UT exams of the four above-the-weld OD circumferential crack indications at ONS-2 (nozzle 18) and ONS-3 (nozzles 23, 50, 56) indicate circumferential extents of approximately 36 degrees, 66 degrees, 59 degrees, and 165 degrees, respectively (see Section 2.2). It is unknown whether each of these cracks grew from a single OD initiation site, or from several initiation sites that linked together to form a long circumferential OD surface crack. Therefore, the initial flaw size used in the Monte Carlo simulation is a shallow semi-elliptical flaw with a circumferential extent uniformly distributed between zero and 180 degrees. Postulating a single flaw with an initially long circumferential extent is an approximation of the possibility of multiple linked initiation sites. The ONS plant experience is consistent with a uniform distribution of initial flaw extent and this approach is reasonable in light of the sparse industry data available for OD flaw distributions. A practical upper limit for this **initial** flaw distribution is a circumferential extent of 180 degrees, which is related to the nature of the stresses on the surface of the CRDM nozzle above the weld. On the nozzle OD above the weld, crack initiation in the circumferential direction may be driven by the axial bending stresses that are related to the proximity of the weld and shrink fit zones, and these are different on the uphill and downhill side of the nozzle. This approach of postulating an initial flaw as long as 180 degrees in circumferential extent bounds the uncertainty from scarcity of probabilistic fracture mechanics data for OD flaw distributions and multiple initiation sites.

Another source of uncertainty is the crack growth rate for OD-initiated PWSCC. The flaw growth rate distribution used in the Monte Carlo simulation is based upon industry data for PWSCC. Parameters affecting growth rate, such as stress intensity and temperature, were distributed in the Monte Carlo model to address uncertainty. Figure 6 illustrates the resulting crack growth rate distribution that was assumed in the Monte Carlo simulation. However, it is unknown whether the difference in environment between the nozzle exterior and interior may affect the growth rate for OD PWSCC. The approach used in this risk assessment to ensure that the uncertainty associated with crack growth rate is bounded, is to benchmark the Monte Carlo simulation results for time-to-TW crack against the plant observations. If the crack growth data are reasonable, the Monte-Carlo simulation should predict TW crack times consistent with the plant observations.

The results of the Monte Carlo simulation for TW cracking are illustrated by the histogram shown in Figure 7. The Monte Carlo simulation results are consistent with the plant experience. Of the 15 leaking CRDM nozzles found at ONS and ANO-1, two had above-the-weld OD circumferential cracks that were TW or almost TW (ONS-3 nozzles 50 and 56). To reach the equivalent percentage of TW cracks in the Monte Carlo simulation (i.e., 13.3% of the samples) required 4.2 years. Anecdotal evidence suggests that some of the nozzles at ONS may have been leaking for as much as 5 to 10 years. Therefore, the results of the Monte Carlo simulation appear to be reasonable or conservative with respect to experience.

The Monte Carlo simulation was used to grow the initial flaws to failure using the crack growth model described in Section 3.3.2 and stress distributions that are characteristic of the most exterior nozzles (i.e., the highest angle of penetration). For this analysis, failure is defined as insufficient ligament to meet ASME Code primary stress limits, which corresponds to a circumferential crack extent of approximately 292 degrees or 81% (Reference 18). The failure definition is conservative since the threshold ligament is based on satisfying primary stress limits using a safety factor of 3 (and 1.5 for emergency and faulted conditions). The failure definition also does not take credit for the Technical Specification required 1 gpm leak detection capability, which as described in Appendix A may occur at a somewhat smaller crack extent depending upon the radial clearance in the penetration annulus. A conservative failure definition is appropriate for this risk assessment considering the current weakness in industry understanding of OD PWSCC and because it may bound uncertainties inherent in the probabilistic fracture mechanics data.

The results of the Monte Carlo simulation are illustrated by the histograms shown in Figures 7 and 8. Based on the Monte Carlo simulation, an OD-initiated crack above the J-groove weld would take a mean time of 8.9 years to grow to a through-wall state, and a mean time of 28 years to result in nozzle failure (or LOCA) due to net-section stress. For comparison, 97.5% of the time-to-failure distribution (non-parametric) is greater than the point estimate reported in Section 3.3.2 (7.5 years for a crack to reach 75% circumferential extent). This reflects the conservatism that is inherent in the deterministic approach.

The time-to-failure histogram (Figure 8) has been partitioned into two-year probability increments to correspond to the worst-case visual inspection intervals for B&WOG plants (i.e., plants with a two-year fuel cycle). The table (see Figure 8 inset) shows the probability that the OD crack will grow to failure within the time indicated assuming there is no detection by visual inspection (boron crystals). The opportunities for detection will be added at two-year intervals in the event tree quantification (see Figure 5).

9.3.5 Probability of Core Damage

The most likely consequence of CRDM nozzle failure (critical size crack) is leakage that is within the capacity of the makeup system. If a complete severance of the CRDM nozzle occurs, the break size will be within the range of what most B&WOG PRAs identify as a medium break LOCA. However, a smaller break size could result if there is a partial failure of the nozzle.

The conditional probability of core damage given a small- or medium-sized LOCA can be readily determined from the plant-specific B&WOG PRAs. In a B&WOG PRA, the conditional core damage probability (CCDP) for a medium break LOCA is on average worse than for a small break LOCA. Therefore, as a representative value, the risk assessment uses the average CCDP for a medium-break LOCA from a survey of the B&WOG PRAs, which is approximately 4×10^{-3} (Reference 35). Use of this CCDP is conservative because plant mitigation response will be better for a break at the top of the vessel than for the LOCAs typically considered in the PRAs (see Section 8.0).

9.3.6 Risk Results for OD PWSCC

The estimated frequency and probabilities in the preceding sections are used to quantify the event tree shown in Figure 5. The event tree shows the progression of sequences starting with the initiating event "CRDM leaks." Each sequence can result in success (e.g., no core damage) or failure/core damage, as noted by the "S" and the "CD" in the "Success or Core Damage" column. One sequence is identified as "CD Residual," recognizing that inspection and crack growth may continue beyond the eight years explicitly modeled in the event tree. The contribution from these residual sequences is not significant. In the event tree, at each decision point (success or failure), the conditional failure probability (as estimated in the previous sections) is shown. Multiplying the appropriate branch failure probabilities results in the frequency of core damage for each sequence. Only sequences that result in core damage are quantified. When summed, the sequence frequencies provide an estimate of the CDF due to OD PWSCC of the CRDM nozzles, which has a mean value of 3.4×10^{-7} per reactor-year.

Uncertainty in these results has been addressed via the use of conservative assumptions and the use of bounding data inputs for the probabilistic fracture mechanics. In particular, bounding assumptions were made for crack initiation time, initial flaw distribution, and multiple crack initiation sites. The crack growth rates used appear to produce results consistent with the plant observations of TW cracks. Other conservatisms include the human error probability for visual inspections, nozzle failure definition, and LOCA mitigation failure probability. Therefore, it is concluded that the CDF results produced by this risk assessment are reasonable in light of the limited industry knowledge base for this failure mechanism.

The estimated core damage frequency (3.4×10^{-7} per reactor-year) compares favorably to the risk acceptance guidelines contained in Regulatory Guide 1.174 (Reference 36) for core damage frequency. Per these guidelines, the risk of operation with potentially undiscovered CRDM nozzle cracks is categorized as "very small." Regulatory Guide 1.174 also has acceptance guidelines for large early release frequency (LERF). The effect of the nozzle cracks upon LERF is insignificant because the containment safeguards systems are not affected by CRDM nozzle cracking. The reactor vessel missile shields preclude consequential damage to the containment building in the unlikely event of CRDM nozzle detachment. No other collateral damage has been identified that may affect containment safeguards systems. Therefore it is concluded that the risk associated with CRDM nozzle cracking at B&WOG plants is small and consistent with the Commission's Safety Goal Policy.

The public health risk associated with the CRDM nozzle cracking is correspondingly very small. For example, the conditional population dose for a medium break LOCA core damage accident at a typical B&WOG plant (ONS) is 1.1E4 person-rem (Reference 37). For the estimated core damage frequency of 3.4×10^{-7} per reactor-year, this corresponds to a public health risk of only 3.7×10^{-4} person-rem/reactor-year, which is insignificant.

According to Regulatory Guide 1.174, risk insights should be considered in an integrated fashion with traditional deterministic evaluations (such as those discussed in Sections 1 through 8). The deterministic and risk evaluations taken together indicate that safety margins and defense-in-depth are not significantly affected by the CRDM nozzle cracking. With effective visual inspections, the nozzle cracking does not significantly increase the LOCA frequency that is assumed in B&WOG PRAs, nor does it increase the frequency above the level that is assumed for design basis accidents. The consequences of a CRDM nozzle failure are less severe than the LOCAs assumed in the FSAR analyses. Also, the CRDM nozzle cracking has no effect on core damage mitigation, containment safeguards, or emergency planning effectiveness. Therefore, this risk analysis concludes that the risk to the public due to CRDM nozzle cracking is acceptable. The risk analysis also supports the findings of the deterministic analyses, which is that visual inspections of the RV head will discover signs of CRDM nozzle leakage before there is a significant likelihood of total failure of a CRDM nozzle due to PWSCC.

10.0 Summary and Conclusions

A safety assessment has been performed to address the potential for PWSCC cracking of RV head penetration nozzles and welds at the B&WOG plants. It addresses both axially and circumferentially oriented flaws that have been observed in the Alloy 600 CRDM nozzles as well as axial/radial flaws observed in the Alloy 182 J-groove partial penetration welds used to attach Alloy 600 CRDM nozzles to low alloy steel RV heads. This safety assessment utilizes and builds upon the existing analyses performed for CRDM nozzle PWSCC (References 3, 7, and 8).

The results of detailed stress analysis of the nozzle and weld regions of the RV head demonstrate that the circumferential, or hoop, stress is generally higher than the axial stress at the same location. On the downhill side of the nozzle, the ratio of hoop stress to axial stress is about 2/1, and on the uphill side it is about 3/2. In the weld region, hoop stresses are about two times the axial stress at the same location. It can therefore be concluded that if PWSCC cracking were to occur, flaws would predominantly be oriented in a longitudinal, or axial, plane, and as such would not promote catastrophic failure of the nozzle by ejection.

Based on laboratory test data for Alloy 182 weld metal in a PWR environment, crack growth through the J-groove weld could occur rapidly (i.e., within one or two years). Although continued crack growth into the low alloy steel head would not be expected due to the low susceptibility of this material to SCC, flaws in the weld metal could continue to grow into the Alloy 600 CRDM nozzle, as seen at ONS-1 and ONS-3. It has been predicted that it would take almost four years for an axial OD nozzle flaw to grow through-wall to the inside surface. At this point, a leak path into the annular region between the nozzle and head could be present, depending on the location of the original flaw in the nozzle.

Any circumferential flaw above the weld on the outside surface of the nozzle should not be considered a safety concern. A short, isolated flaw would take more than 10 years to grow through-wall, while a long circumferential (where multiple flaws have joined) could grow from the outside surface to the inside surface in about 3.5 years. In neither case would the structural integrity of the nozzle be compromised to the point that the nozzle would fail by ejection.

Circumferential cracking has also been observed on the outside surface of CRDM nozzles at ONS-3, at the toe of the fillet weld that forms part of the structural attachment to the reactor vessel head. Since these cracks are located at or below the weld, and not in the reactor coolant pressure boundary, they are not considered to be a safety concern from the standpoint of gross structural failure or release of radioactive water. Due to the proximity of associated through-wall cracking below the weld, however, there is a concern that a through-wall circumferential crack could link up with two or more through-wall axial cracks and form a loose part.

Based on experience at ONS-3, circumferential and axial cracking below the weld is accompanied by through-wall axial cracking at and above the weld, as evidenced by deposits of boric acid crystals on the top of the head. It is concluded from these results and observations that detectable leakage would precede the development of a loose part.

Concerns relating to a lack of fusion type weld defect between the nozzle and weld have been addressed by considering the growth of a postulated "wrap-around" circumferential flaw along the cylindrical surface at the nozzle-to-weld interface. Utilizing radial stresses between the nozzle and weld and PWSCC crack growth rates for Alloy 182 weld metal, it has been calculated that it would take two years for a 17.5% wrap-around flaw to grow to an allowable 67% flaw size.

It has also been demonstrated by a detailed stress analysis that annular gaps develop between the CRDM nozzle and the RV head in the RV head penetration of the B&WOG plants. In the event of a through-wall crack in the J-groove weld or the portion of the CRDM nozzle in the annulus, these gaps form the natural leakage path for the RCS coolant to the OD of the RV head. Assuming a designed 0.0010 inch nominal diametric interference, the minimum calculated radial gap is 0.001 inch for both the center nozzle and the outermost nozzle designs. The average or representative radial gaps for the center nozzle and the outermost nozzle are 0.0016 inch, and 0.002 inch, respectively.

Axial flaws are anticipated to be predominant at both the ID and OD of the CRDM nozzle based on the magnitude of the hoop stresses, although circumferential flaws can be envisioned on the OD and have been observed (both above and below the weld). Axial flaws within the J-groove weld are also the most plausible flaws due to high hoop stresses. These types of cracks are envisioned to break the surface as pinhole type cracks or as tight PWSCC cracks. These tight cracks would result in very low leakage rates as evidenced by the low volume of boric acid crystals found in the vicinity of CRDM nozzles at ONS-1, ONS-2, ONS-3, and ANO-1. It was estimated that approximately 0.5 in^3 was present around CRDM nozzle number 21 at ONS-1. However, observable leakage is expected to occur well before crack propagation would reach ASME Code limits.

It has been shown that, assuming a large portion of the nozzle cross-section contains a through-wall circumferential crack, there is ample room for leakage to occur before approaching the net section limit ligament. This will allow a detectable leakage of steam through this large crack, thereby providing ample warning to prevent the failure of the nozzle. In addition, evidence indicates that the nozzles are in an oval shape due to interaction with the closure head deformation. Therefore, there are gaps between the nozzle and the head that will provide sufficient leak paths for a fairly large volume of steam to escape thereby providing leak detection.

The allowable lack of fusion size was previously determined to be 67%, or 8.4 inches of circumferential extent. Also, the critical lack of fusion size was determined to be 85% or 10.6 inches of the circumference. The leakage rates were predicted using the same methodology as used for the evaluation of the axial crack. It was determined that a crack length of 7.5 inches is required for the center nozzle design to achieve a 1 gpm leak rate. Similarly, it was established that a crack length of 5.0 inches is required for the outermost nozzle design to achieve a leak rate of 1 gpm. Since these cracks are less than the allowable lack of fusion crack length of 8.4 inches, it is concluded that these types of cracks will be detected by the plant's leak detection capability.

Boric acid corrosion concerns were addressed for a variety of conditions and leakage rates potentially assumed to occur. It was determined that corrosion of the RV head penetration, at a maximum volumetric metal loss rate of 1.07 in^3/yr would be possible. Various defect profiles were postulated to model this level of corrosion for a time period of six years. It was concluded that safe operation of the plant would not be affected as a result of this level of corrosion and that within this time, the leak will be detected during a walk-down inspection of the RV head area.

All of the observed through-wall CRDM cracks in the B&WOG plants have been traced to origination in the vicinity of the weld and not at the end of CRDM nozzle. Failures in the end of the nozzle have the potential to generate loose parts that could relocate within the RCS and compromise equipment operation or fuel-clad barrier integrity. Given the current knowledge of the residual stresses in the CRDM nozzles, FRA-ANP has concluded that the through-wall axial cracks present below the weld initiate at the toe of the weld. These cracks are not expected to propagate to the point that a loose part will be generated before some leakage is visible. Therefore, all aspects of the CRDM cracks have been considered from a safety analysis perspective. This review has concluded that simultaneous multiple CRDM nozzles will not fail and that the failure of a single CRDM nozzle is bounded by both the LOCA and non-LOCA plant analyses already completed to support current plant operation.

A loose part evaluation was performed to evaluate the potential for loose parts from a failed CRDM nozzle to potentially enter a control rod guide tube and prevent the control rod assembly from being fully inserted. It was concluded that there was at least a 25 percent chance of a loose part entering the guide tube and potentially impairing successful operation of that assembly. The LOCA and non-LOCA analyses assume that the control rod of highest worth is stuck out of the core. In addition, only a fraction of the remaining worth is used in demonstrating that at least a 1 percent shutdown margin exists at hot zero power conditions.

It has been demonstrated through risk analysis that the risk from potentially undetected CRDM nozzle cracks is "very small" per the guidelines of Regulatory Guide 1.174. The estimated core damage frequency due to OD PWSCC of the CRDM nozzles is 3.4×10^{-7} per reactor-year. Conservative assumptions are made in the risk assessment to address uncertainty in the estimates of human reliability, probabilistic fracture mechanics, and plant mitigation response. Taken together with the results of the deterministic analyses, the risk analysis demonstrates that visual inspections of the reactor vessel head will be sufficient to minimize public risk. The visual inspections will discover signs of CRDM nozzle or penetration weld leakage before there is a significant likelihood that the leakage will cause CRDM nozzle structural failure or detachment due to outside diameter PWSCC.

Finally, all evidence to date suggests that it will require several years for the material to degrade to the point that total failure of the component could occur. During that time, if a crack should form, leakage of primary coolant on to the RV head can be identified through routine visual inspections. The component can then be repaired and returned to service without jeopardizing the health and safety of the public.

As a result of the previously described activities and evaluations performed by the B&WOG, the following conclusions have been reached regarding degradation of CRDM nozzles, thermocouple nozzles, and RV head attachment welds at B&WOG plants:

1) The B&WOG plant safety evaluation (Reference 3) remains valid.

2) The B&WOG utilities comply with 10CFR50.55a and continue to meet the intent of General Design Criteria 14, 30, 31, and 32 of Appendix A of 10CFR50.

3) The potential for the B&WOG plants to have sulfur-induced IGA or SCC of CRDM and thermocouple nozzles is very low (Reference 5).

4) The risk to the public due to CRDM nozzle cracking is "very small" and acceptable per the guidelines of Regulatory Guide 1.174.

5) Visual inspections of the reactor vessel head will discover signs of CRDM nozzle leakage before there is a significant likelihood of total failure of a CRDM nozzle due to PWSCC.

6) Inspections, other than visual examinations in accordance with GL 88-05, are not necessary from a safety perspective.

7) One of the most susceptible B&WOG plants, ONS-2, has inspected all 69 CRDM nozzles in 1994 and two follow-up inspections on the nozzles identified with flaw-like indications. Recent observations at ONS-1, ONS-2, ONS-3, and ANO-1 have also added credence to the safety assessments that have been performed.

8) All B&WOG utilities continue to perform visual inspections of the RV head in accordance with their respective Generic Letter 88-05 and Bulletin 2001-01 responses.

9) The B&WOG will continue to share B&WOG plant inspection data and participate in agreed upon joint Owners Group (e.g., MRP) activities with the U.S. nuclear industry on this issue.

10) The B&WOG will continue to monitor this issue.

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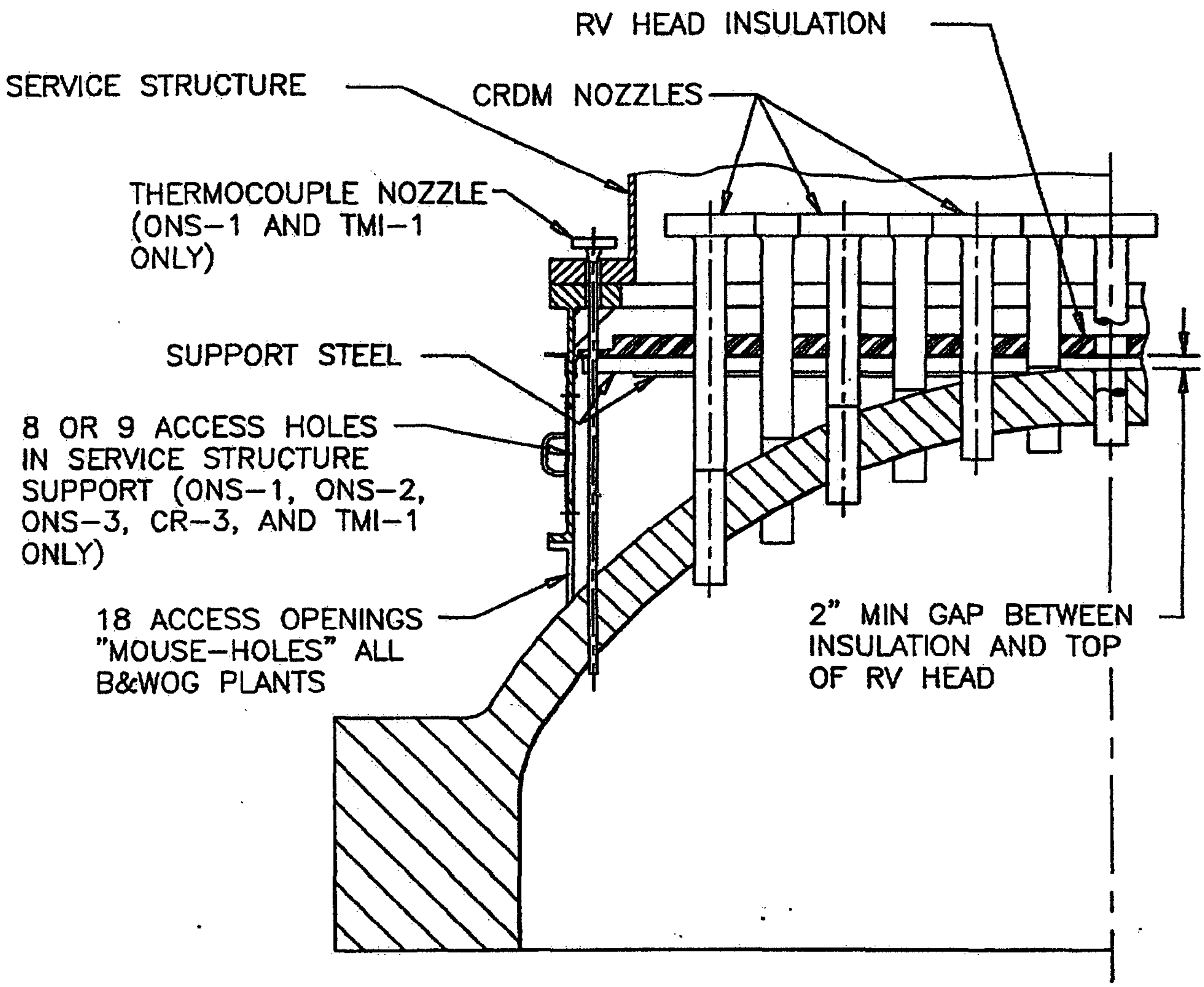
Figure 1. Side View Schematic of B&W-Design Reactor Vessel Head, CRDM Nozzles, Thermocouple Nozzles, and Insulation.



Note: The thermocouple nozzles were removed from ONS-1 at EOC-19.

Figure 2. Plenum Cover Assembly.

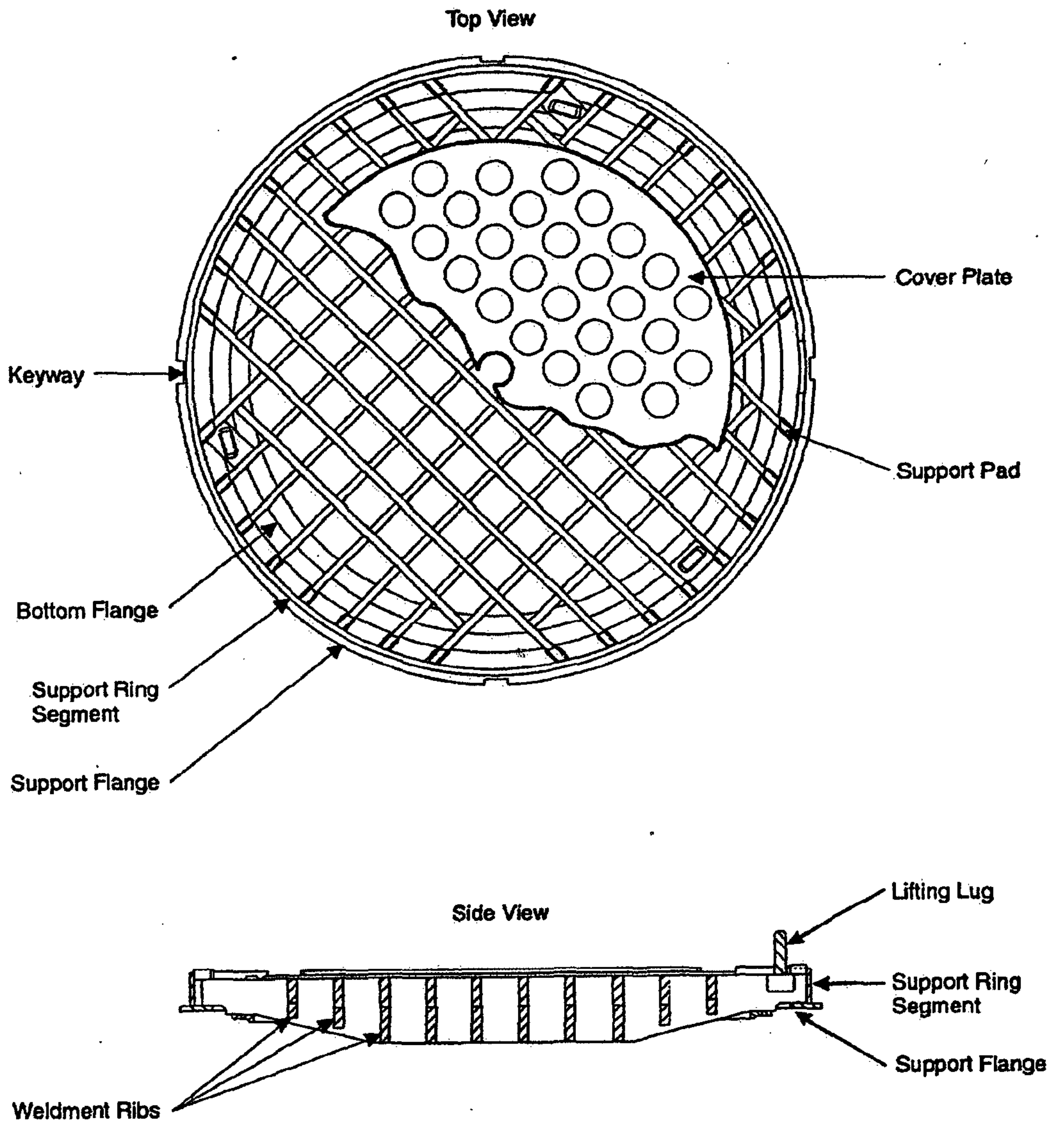


Figure 3. Control Rod Spider Assembly.

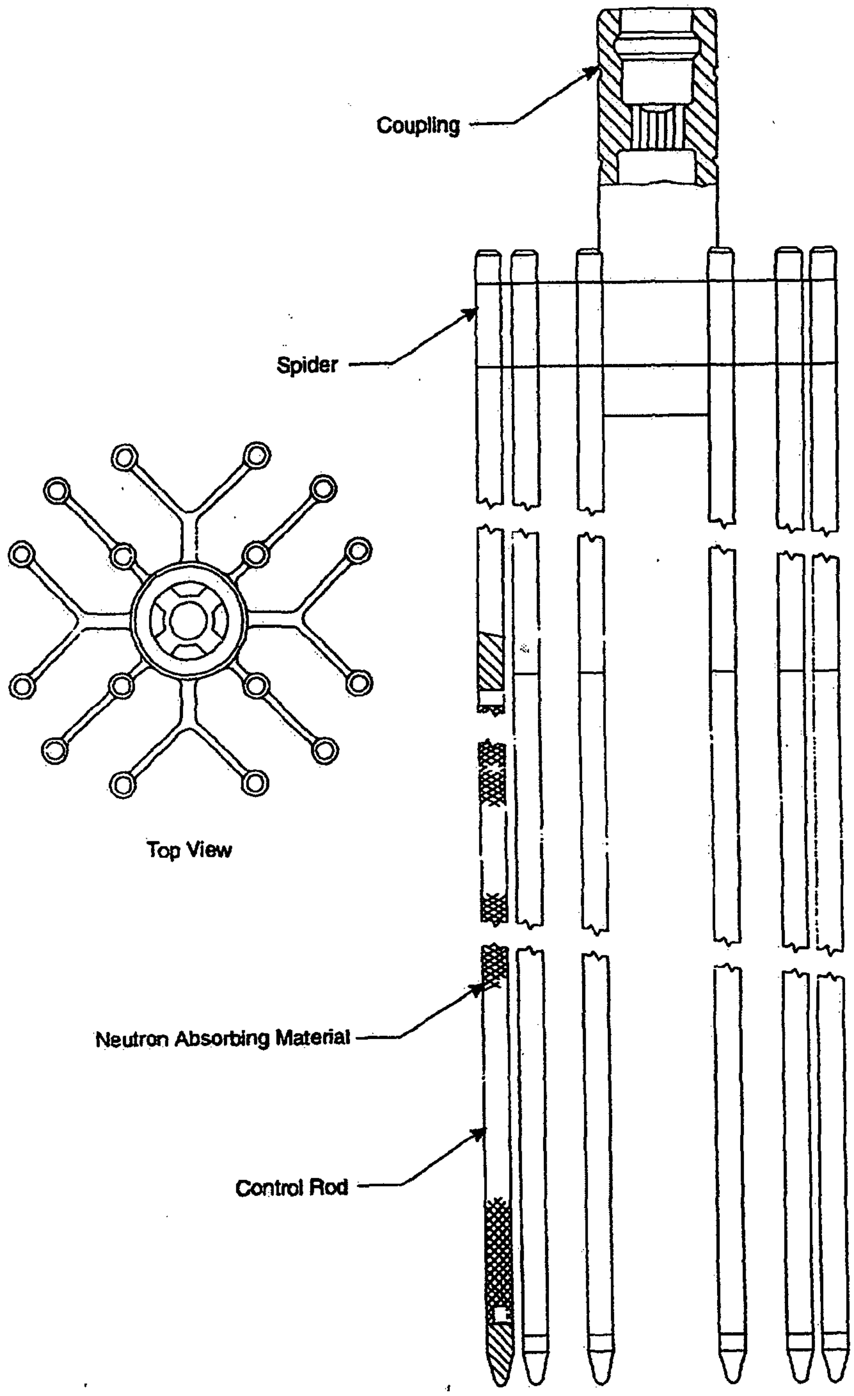


Figure 4. Control Rod Guide Brazement Assembly

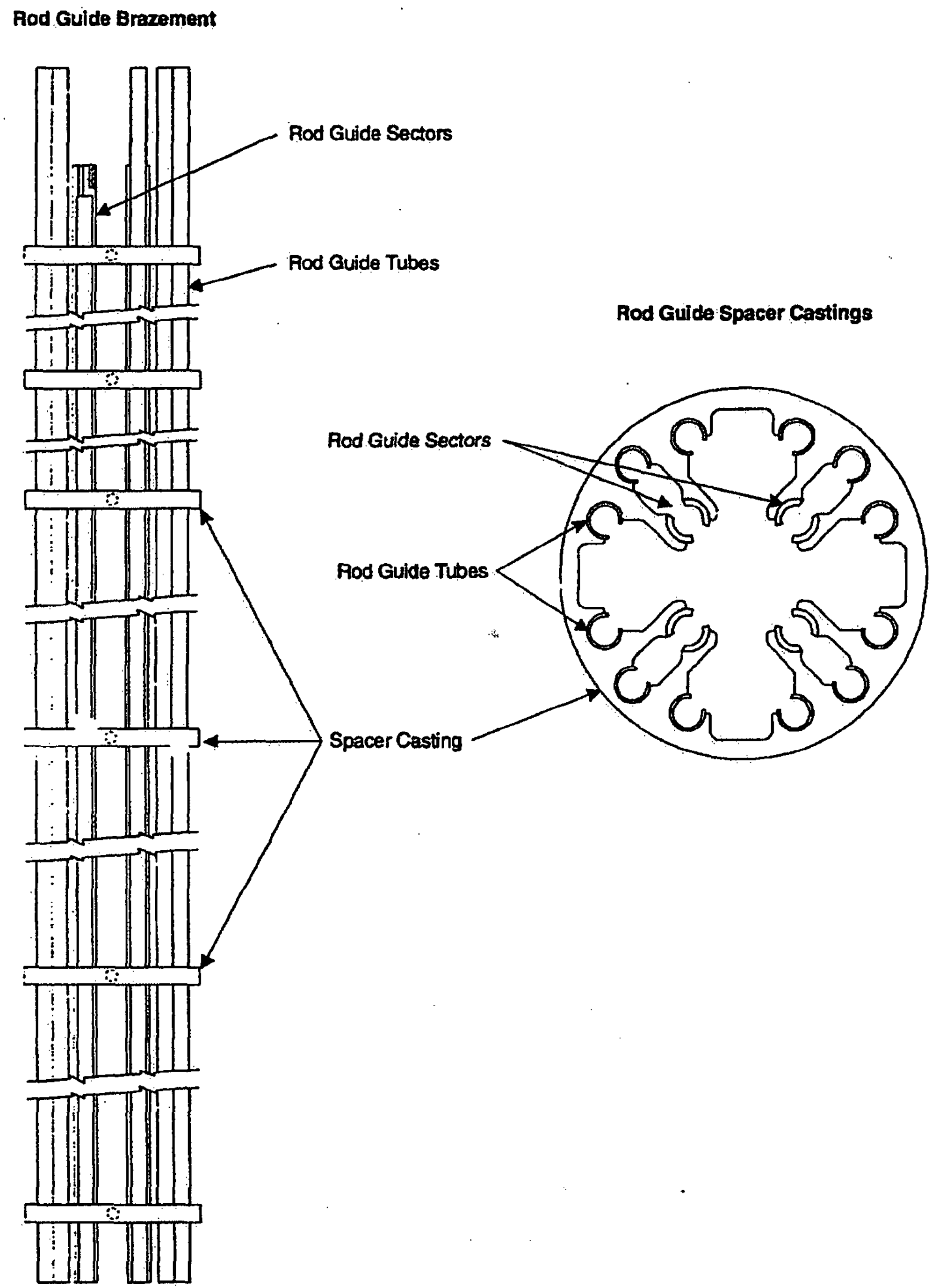


Figure 5. Event Tree for Frequency of Core Damage from Outside Diameter PWSCC in B&WOG CRDM Nozzle

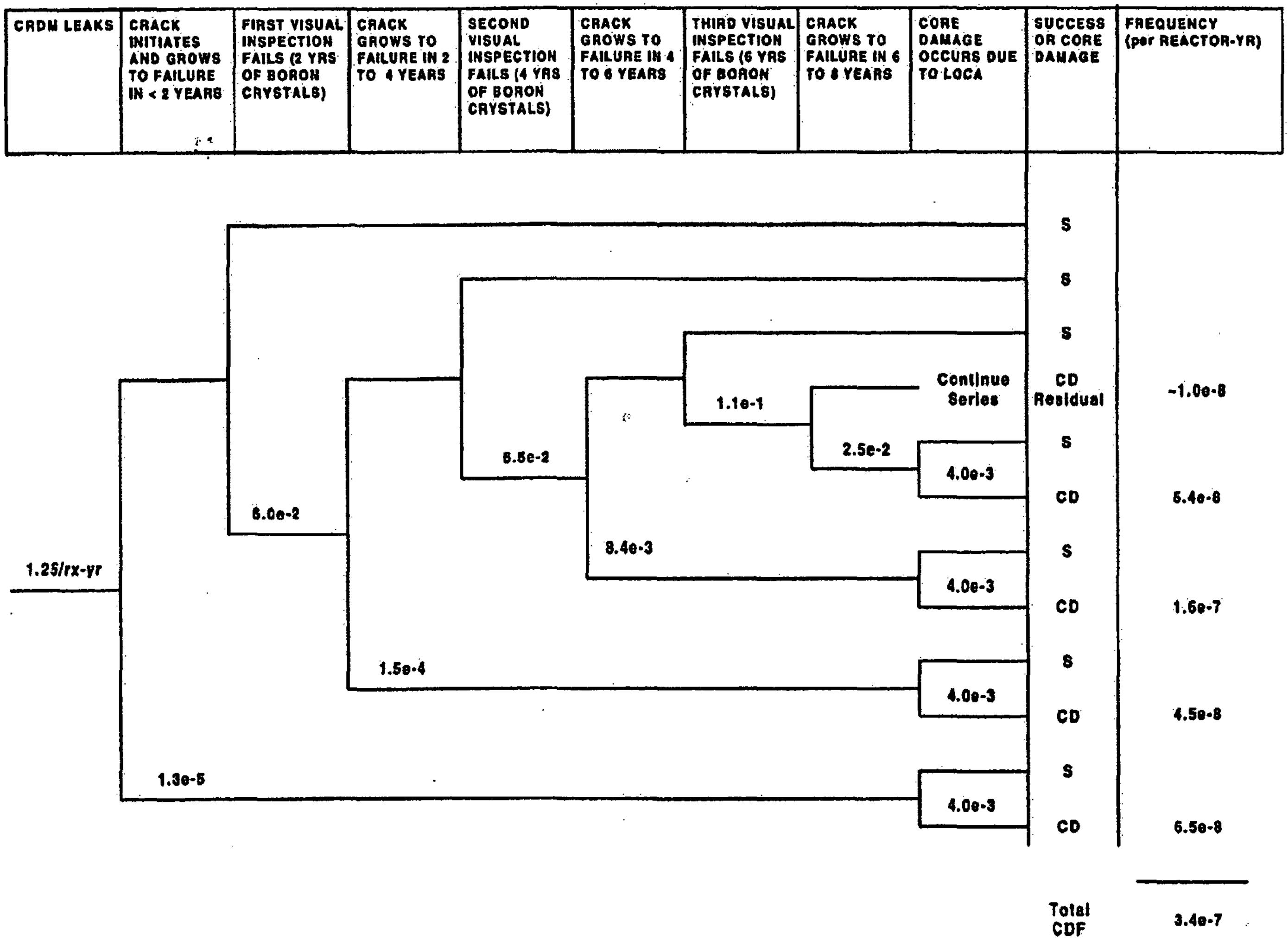


Figure 6. Crack Growth Rate Assumed in Monte Carlo Simulation

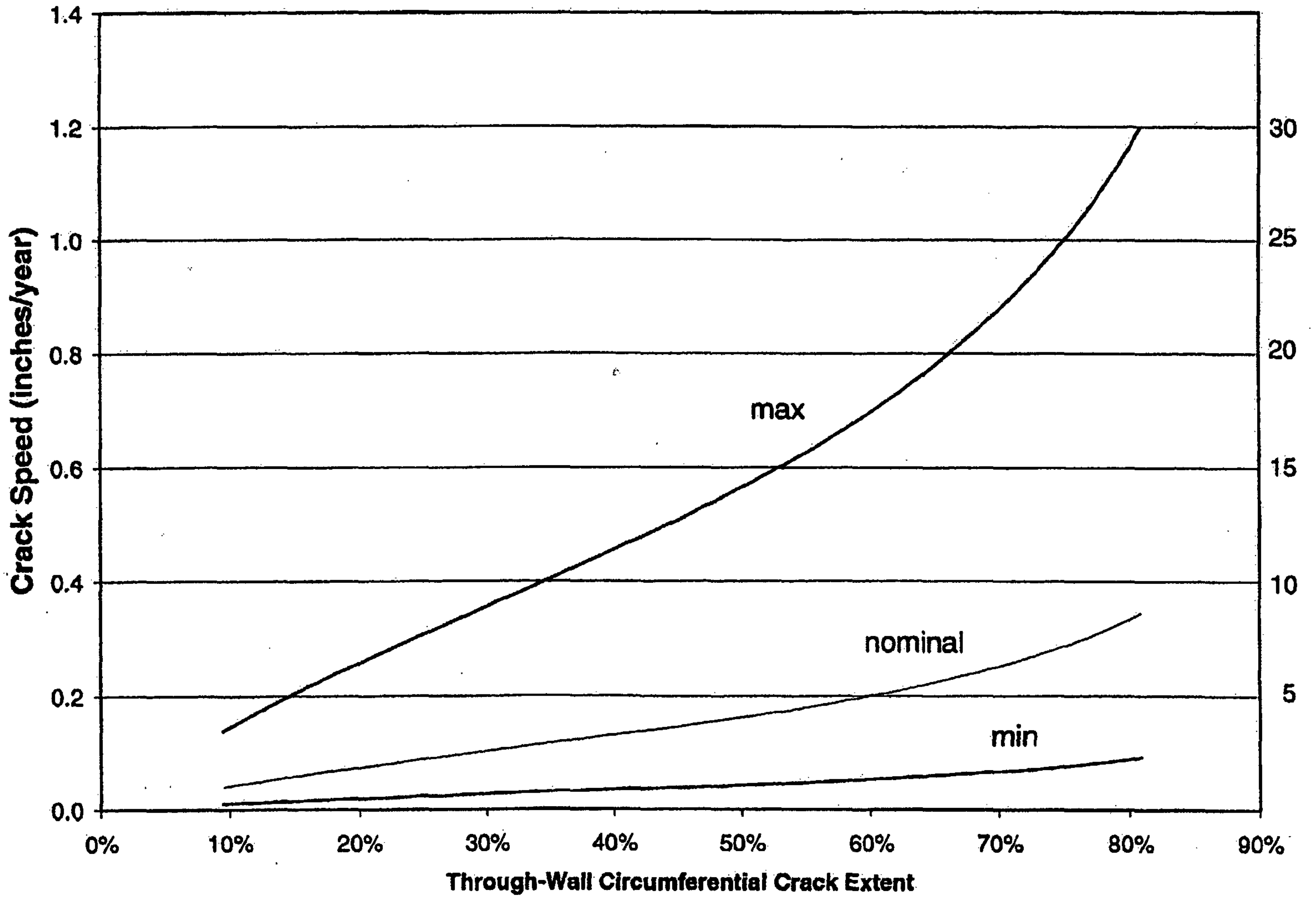


Figure 7. Probability of Through-Wall Crack versus Time after Initiation of Outside Diameter PWSCC

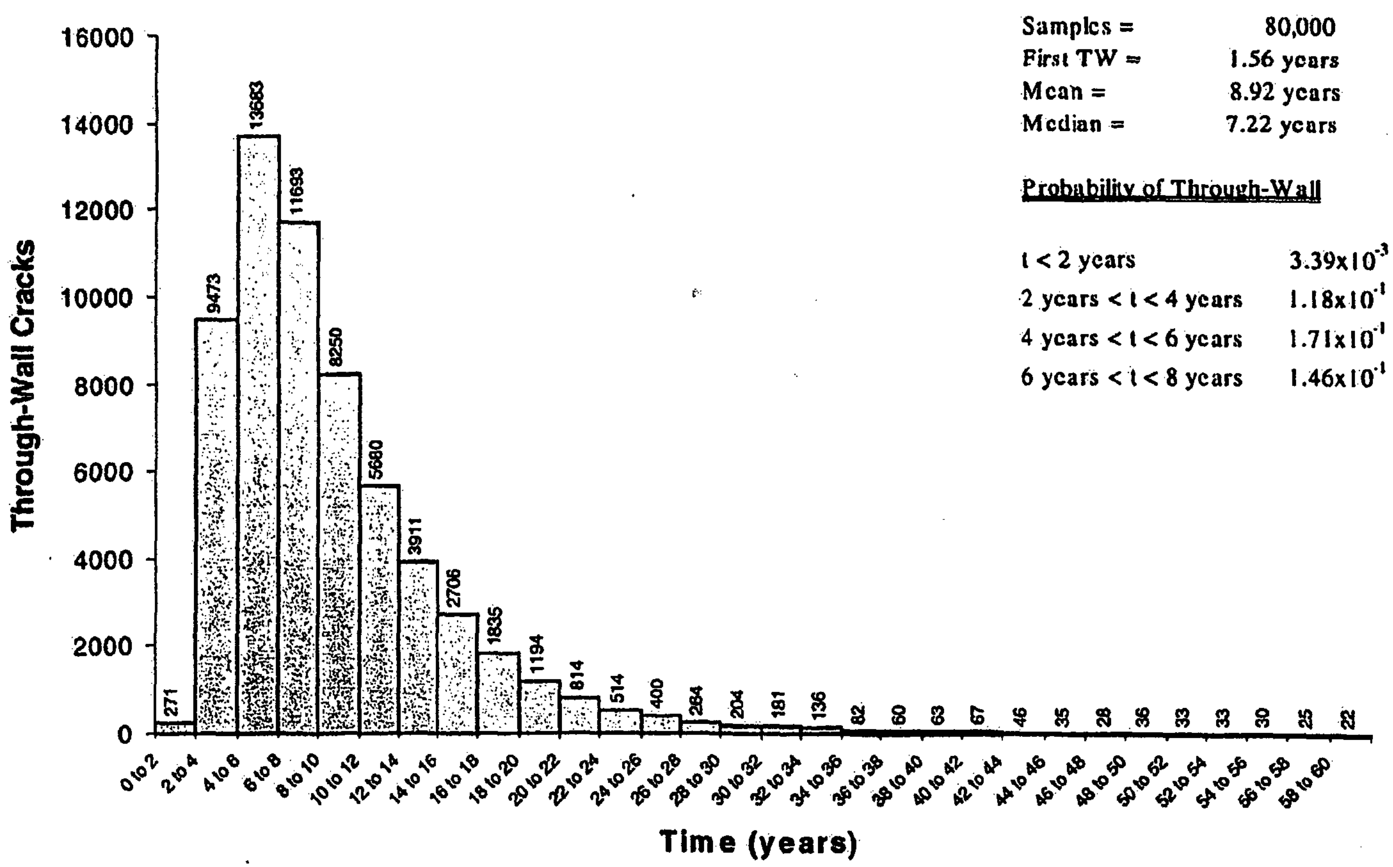
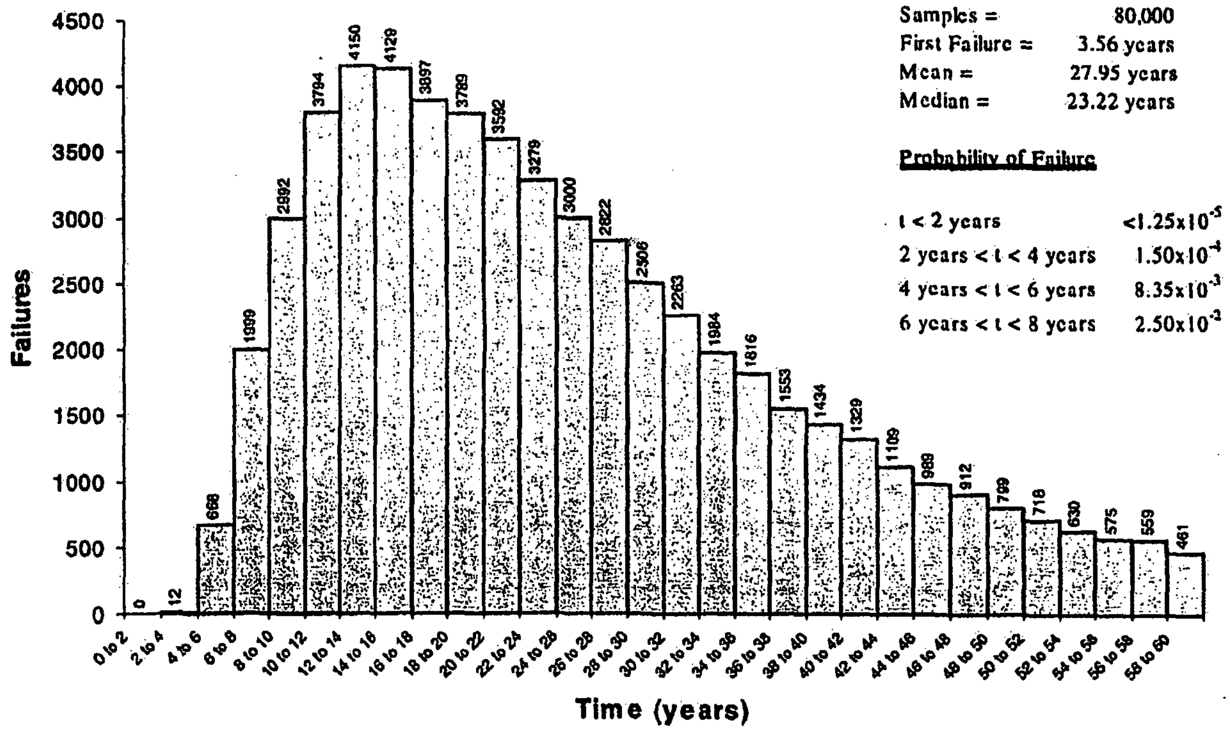


Figure 8. Probability of Net-Section Failure versus Time after Initiation of Outside Diameter PWSCC



Appendix A

Leakage Assessments

The leakage assessments due to various postulated flaws and due to lack of fusion in the CRDM nozzle/J-groove weld region are addressed in this section. As a result of stress analyses of the B&W-design CRDM nozzles, it has been previously demonstrated that during normal operation steady state conditions an annular gap develops (above the CRDM weld to the RV head) between the CRDM nozzle and the RV head penetration (Reference 3). Of particular interest is the prediction of a radial gap in a previously interference-fit region. The prediction of this radial gap during steady state operating conditions is utilized in the assessment of leakage rates through the CRDM nozzle/head annulus.

The radial gaps are different for the two types of CRDM nozzles that were evaluated by a detailed stress analysis, the center nozzle design and the outermost nozzle design. For the center nozzle, the initial interference-fit between the nozzle and the head separates to form a 0.003 inch maximum radial gap above the weld during steady state conditions. The average radial gap is 0.0016 inch and the minimum radial gap is 0.001 inch as illustrated in Figure A-1. For the outermost nozzle, the radial clearance in the initial interference fit region is approximately 0.001-inch minimum during steady state conditions as depicted in Figure A-2. However, a major portion of the periphery of the CRDM nozzle/RV head penetration shows a radial clearance of at least 0.002 inch and the maximum radial gap is about 0.003 inch.

A.1 Axial Flaws in CRDM Nozzle Above the J-Groove Weld

Leakage assessments for postulated through-wall axial flaws in the CRDM nozzle above the J-groove weld were previously addressed in Reference 3 and summarized below.

Reactor coolant system (RCS) leakage rates through postulated CRDM nozzle cracks and the annulus clearances between the nozzle and reactor vessel head were predicted by a parametric analysis. Both the crack lengths and annulus clearances were varied. Because of the high pressure-high energy conditions in the RCS, the sub-cooled coolant saturates, flashes, and then chokes at the exit of either the crack or annulus.

Leakage rates were obtained through an iterative process using the Homogeneous Equilibrium Model (HEM) critical flow tables and by solving single and two-phase pressure loss correlations. Since the flow chokes at either the exit of the crack (i.e., crack/annulus interface) or at the exit of the annulus (i.e., top of the penetration) for any given crack length and annulus clearance, both possibilities were considered in the analysis.

Therefore, the flow through the crack and annulus clearance was broken into two separate leakage flow paths to account for the two possibilities: (1) single and two-phase flow through the crack with choking at the exit of the crack, and (2) single phase flow through the crack and single and two-phase flow through the clearance annulus, with choking at the exit of the annulus.

For the first path with flow choking at the exit of the crack, the downstream leakage paths were calculated. The path with the lesser flow rate was considered to have the actual flow rate. Because of the choking properties of the flow, the greater flow rate was not possible. Thus, if the flow rate through the path with choking at the exit of the crack is less than that through the crack and the annulus, then the flow rate through the crack and the annulus is limited by choking at the exit of the crack.

In crack limited problems, the flow chokes at the crack exit. The pressure just upstream of the exit is assumed to be the exit pressure. Using this pressure, the RCS enthalpy, and the HEM tables, a trial critical mass flux is established. This flow rate is used in crack pressure loss calculations to determine a new value for the exit pressure. When the assumed and calculated values of the exit pressure agree, the solution has converged and the crack limited flow rate is established. The crack pressure loss calculations are divided into two calculations: sub-cooled flow and two-phase flow.

The results of the analysis show that for annulus clearances greater than 0.0001 inch and crack lengths less than 3 inches, the limiting factor is the size of the crack, while in cracks longer than 3 inches, the flow does not reach saturation conditions in the crack and therefore chokes at the exit of the annulus. For an annulus clearance less than 0.0008 inch, the flow rate will not exceed 1 gpm regardless of crack size. Likewise, for a crack length of 2 inches and shorter, the leakage flow rate will not exceed 1 gpm regardless of annulus clearance.

For a crack length of 2 inches and a maximum annulus clearance of 0.003 inch, the leakage flow rate was determined to be 0.559 gpm. However, it was demonstrated that as the crack extends from 2 to 3 inches in length, the flow rate would approach and exceed the leak detection capability rate of 1 gpm for annulus clearances of 0.001 inch and greater.

In addition, an independent leakage assessment was also performed as documented in Reference 38 and summarized below.

The objective of the report was to demonstrate that sufficient leakage of primary coolant, beyond the 1 gpm leak detection capability per Regulatory Guide 1.45 requirements, is feasible if a PWSCC indication of sufficient size occurs in the CRDM nozzle. The evaluation was based on applicable industry leak test data to the CRDM nozzle/head annulus (subsequently written as "CRDM annulus") gap. An inventory of experimental data on two-phase critical flow experiments were reviewed to help identify those that are applicable to the problem of predicting leakage rates through the CRDM nozzle and the annulus between the nozzle and the RV penetration.

Only the most pertinent data from the literature of experimental investigations were considered in the assessment of leakage rate through the CRDM annulus. The experiments were determined to be pertinent based on review against key thermal-hydraulic parameters for the evaluation of leakage through the CRDM nozzle/closure head annulus.

The pertinent data were identified in the work of Agostinelli, et al., Amos and Schrock, and Matsushima, et al. (see Reference 38 for these citings). The data from the first two references, when related to the CRDM problem predicted leakage rates greater than 1 gpm. The data from the third reference, when related to the CRDM problem, corresponded to a leakage rate of 0.6 gpm. However, the experiment was based on a stagnation pressure of only 975 psi and the stagnation pressure associated with the CRDM nozzle is 2250 psi. Accounting for the higher stagnation pressure should result in a predicted leakage rate greater than 1 gpm. Therefore, it is concluded in the report that, based on the plant's leak detection capability of 1 gpm within an hour per Regulatory Guide 1.45, the leakage through the CRDM annulus (under the conditions discussed in the report) will be detectable. Furthermore, it should be noted that the prediction of the leak rates given above were conservatively determined using the crack opening area of the CRDM annulus corresponding to a radial gap of only 1 mil. The report also concludes that, should a CRDM nozzle have a through-wall crack, a leak rate of 0.04 gpm to less than 1 gpm will result in significant accumulation of boric acid crystals.

A.2 Axial Flaws Within the J-Groove Weld

Flaws should grow axially through the J-groove weld due to the nature of the stresses in the J-groove weld. For a PWSCC-type crack, it may break the surface as a very tight or pinhole-type crack in the annulus region. These types of cracks would result in a low leakage as has, for example, been observed during the visual inspection of CRDM nozzle number 21 at ONS-1 in December 2000 (Reference 17) and at ONS-3 in February 2001. The maximum amount of boric acid crystals observed around the base of the ONS-1 CRDM nozzle number 21 was approximately 0.5 in^3, signifying a very low leakage rate through the crack. Only small quantities of boric acid crystals were present on the ONS-2, ONS-3, and ANO-1 RV heads, as well.

A.3 External Circumferential Flaw in CRDM Nozzle

An assessment of external circumferential crack growth in the CRDM nozzle above the J-groove weld was addressed in Reference 7. If it is postulated that a circumferential crack propagates through-wall and grows circumferentially along the weld-nozzle interface region, the potential safety concern is detachment of the upper nozzle from the lower nozzle section and its ejection from the closure head. However, detection of leakage prior to tube failure is predicted to occur.

Based on a limit load analysis of the CRDM nozzle geometry, the net section limit ligament is less than 25%. Postulating that a large portion of the nozzle cross-section contains a through-wall circumferential crack, there is ample room for leakage to occur before approaching the net section limit ligament. This will allow sufficient leakage of steam through this large crack to be detectable, thereby providing ample warning to prevent the failure of the nozzle. The flow rates were predicted (without consideration of potential "leak-plugging" in a narrow annulus) for a six-inch circumferential through-wall crack (nearly 50% of the circumferential extent, as observed in nozzle number 56 at ONS-3). For annulus clearances of 0.001 inch, 0.0016 inch and 0.002 inch (to cover the ranges of the predicted clearances during normal steady state operation for the center nozzle to the outermost nozzle), the leakage rates were determined to be 0.4 gpm, 0.8 gpm and 1.2 gpm, respectively.

A.4 Lack of Weld Fusion Areas in the J-Groove Weld

The allowable lack of weld fusion areas in the J-groove weld of the B&WOG plants was addressed in Reference 8. Framatome ANP performed an inspection of the nozzle-to-vessel head welds in a section of Midland Unit 1, which is typical of the B&WOG plants. The inspections revealed that the majority of the indications were located at the CRDM nozzle-to-weld interface, and all indications were less than 2 inches (51 mm) long. Most of the indications detected in the Midland welds are believed to be slag inclusions, with a fewer number of areas indicating lack of fusion of the weld zone. The two areas of concern for the lack of fusion are the CRDM nozzle-to-weld interface and the head-to-weld interface. Both these areas were evaluated to determine the minimum weld area required to meet the ASME Code primary shear stress limits (i.e., allowable lack of fusion size) and to determine the weld area required to limit the shear stress to the shear flow stress (i.e., critical lack of fusion size). It was demonstrated that the CRDM nozzle-to-weld interface was more limiting. The results showed that approximately 67% (corresponding to 8.4 inches of circumferential extent) of the total weld area may be unfused and still meet the ASME Code shear stress limit. Similarly, using the Tresca shear flow stress criteria, it was shown that 85% (corresponding to 10.6 inches of circumferential extent) of the total weld area may be unfused and still have sufficient strength to prevent a catastrophic failure.

The allowable lack of fusion size and indeed the critical lack of fusion size have significant circumferential crack lengths such that sufficient leakage can be demonstrated for these types of cracks.

A leakage assessment for this postulated circumferential crack in the weld was performed using the methodology very similar to that described in Section A.1. The only difference is that only one leakage path is considered which represents the annulus. The flow is assumed to choke at the exit of the annulus.

The crack lengths required to achieve the leak detection capability rate of 1 gpm were determined for annulus clearances of 0.0016 inches and 0.002 inches. These annulus clearances correspond to the average radial gaps during steady state normal operating conditions for the center and outermost CRDM nozzles, respectively. It was determined that a crack length of 7.5 inches in the center nozzle (annulus of 0.0016 inch) is required to achieve a 1 gpm leak rate. Similarly, it was determined that a crack length of 5.0 inches in the outermost nozzle (annulus of 0.002 inch) is required to achieve a 1 gpm leak rate.

Figure A-1. Radial Clearance for Center Nozzle

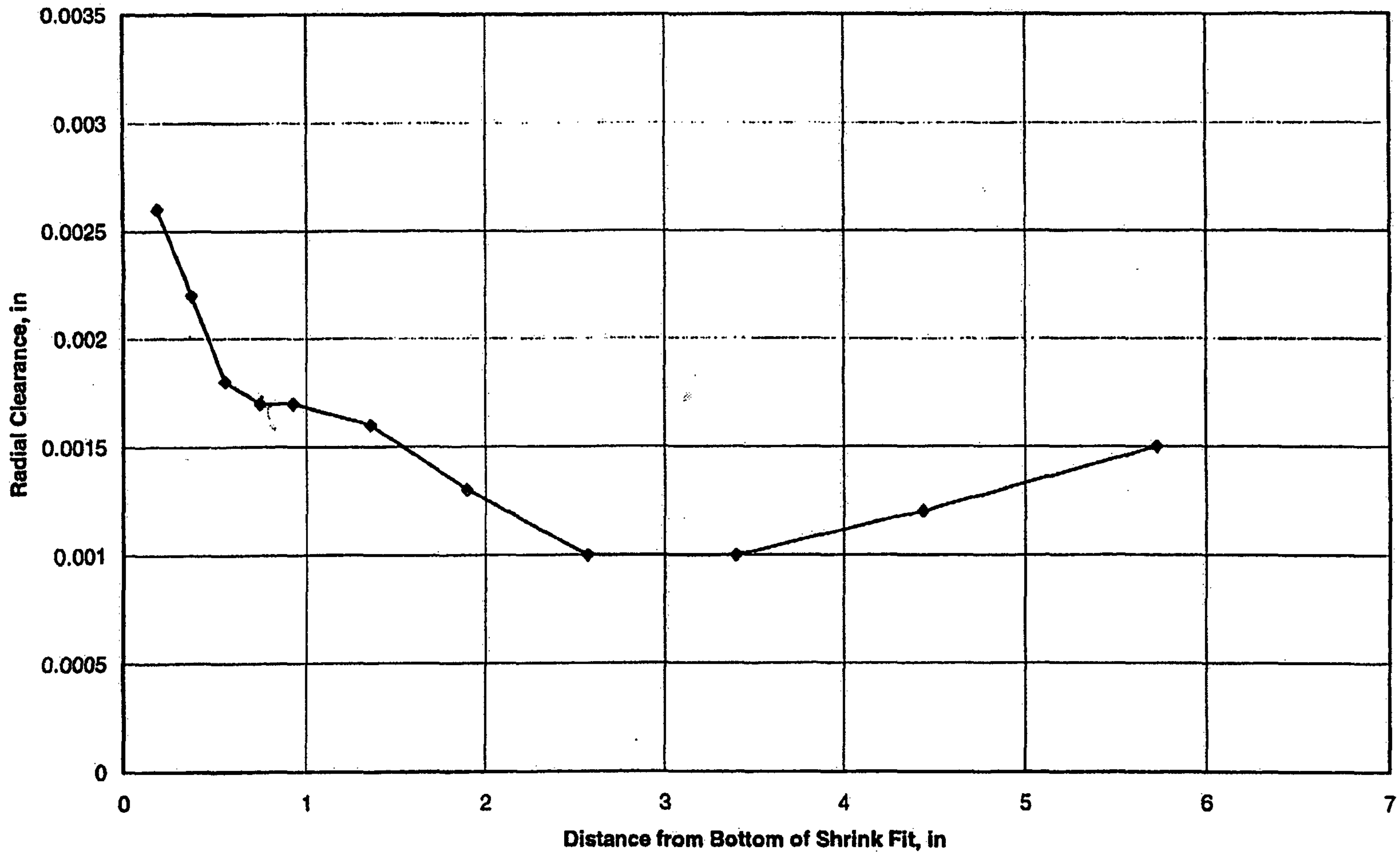


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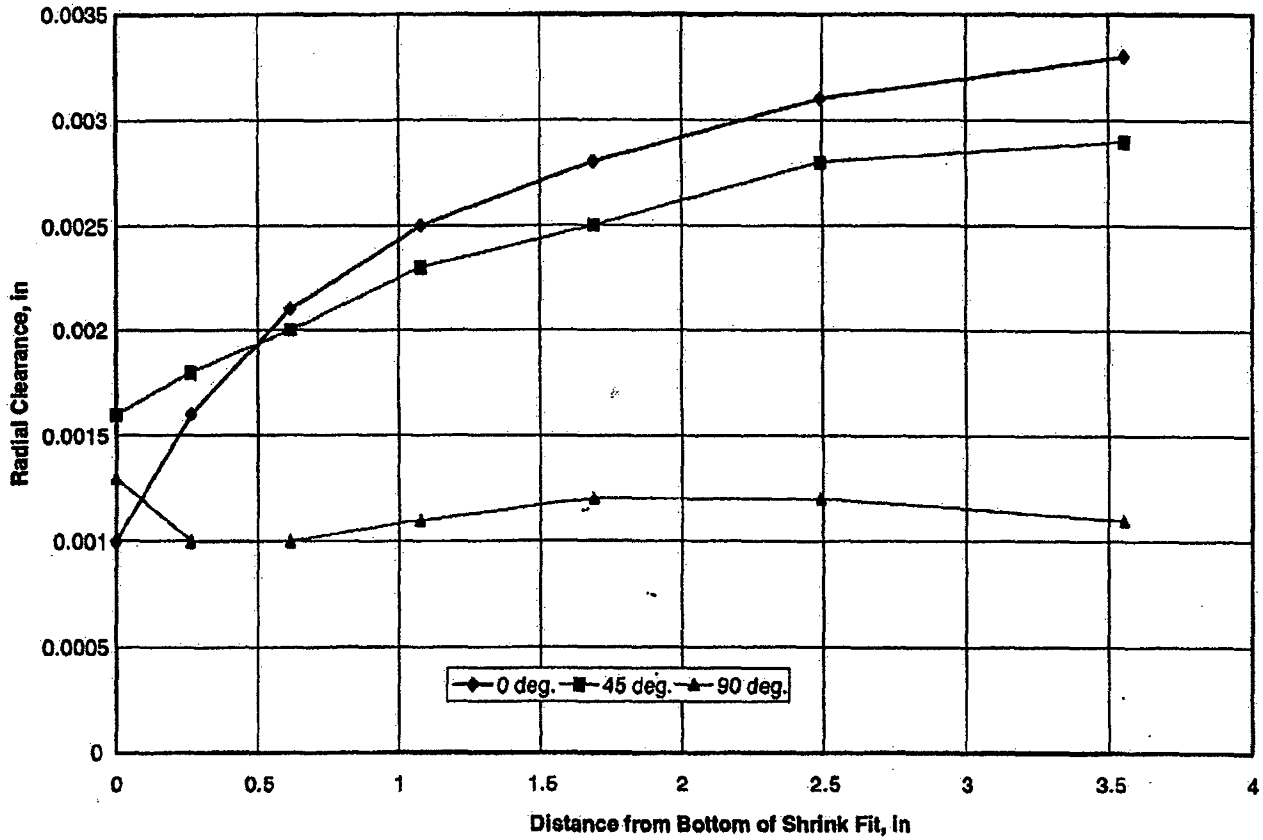
Figure A-2. Radial Clearance for Outermost Nozzle



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Replacement Attachment 5 for Letter Serial Number 2735

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Proprietary Version of
Structural Integrity Associates, Inc. Calculation, File Number W-ENTP-11Q-306, "Finite Element Gap Analysis of CRDM Penetrations (Davis-Besse),"

STRUCTURAL INTEGRITY Associates, Inc.	CALCULATION PACKAGE	FILE No: W-ENTP-11Q-306 PROJECT No: W-ENTP-11Q

PROJECT NAME: Davis-Besse Technical Response to NRC Bulletin 2001-01

CLIENT: First Energy

CALCULATION TITLE: Finite Element Gap Analysis of CRDM Penetrations (Davis-Besse)

PROBLEM STATEMENT OR OBJECTIVE OF THE CALCULATION:

Develop a finite element model of the top head and CRDM penetrations for Davis-Besse Nuclear Power Station. The model is then used to evaluate the gaps between the CRDM tubes and the hemispherical head during normal operating conditions.

Document Revision	Affected Pages	Revision Description	Project Mgr. Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1 – 24 A1 – A3 B1 – B2 C1 – C6 D1 – D3 Project CD-Rom	Original Issue		

1.0 Problem

Develop a finite element model of the top head and CRDM penetrations for Davis-Besse Nuclear Power Station. The model is then used to evaluate the gaps between the CRDM tubes and the hemispherical head during normal operating conditions.

2.0 Finite Element Model

A finite element model has been constructed using the ANSYS finite element software package [1]. The model includes the upper hemispherical head, the upper closure flange and the CRDM housing tubes. Due to the symmetrical nature of upper head structure and the layout of the CRDM tubes, only 45° of the total circumference was modeled. Additional details are described in the following sections. The resulting model can be seen in Figure 1.

2.1 Hemispherical Head/Upper Closure Flange

References 2 and 3 provided the closure flange an'. nemispherical head dimensions used in the finite element model. The flange and head were c.nstructed using the ANSYS 8-node SOLID45 elements. The following assumptions were made during the construction of the segment of the finite element model:

- The clad material was included as base metal for determination of dimensions and modeling.
- The clad was assumed to be a constant $^3/_{16}$ inches thick throughout the structure.
- The bottom face of the closure flange does not specifically model the contact surface. The bottom face remains plain and the contact surface will be simulated via gap elements (described later in the loads section of this calculation).
- The hemispherical head to closure flange fillet radius on the outside surface was assumed to be $6^1/_4$ inches.
- The closure bolt holes were not specifically modeled, thus the closure flange is a solid structure. However, the locations of the bolt holes (there were 7-1/2 holes in the 45° segment modeled) were modeled to provide loading points for the bolt preload.
- Some additional assumptions/variations in the hemispherical head will be described in the following section of the CRDM housings.

See Figure 2 for the dimensions used for the hemispherical head and closure flange.

2.2 CRDM Housing Tube Penetrations

A total of 13 CRDM housing tube penetrations where modeled. They also were modeled using the ANSYS 8-node SOLID45 element. Based on the 45° section modeled, the following tube configurations were actually included:

- 1 is modeled as 45° (top dead center) (Tube 1)
- 7 are modeled as 180° (along the symmetry boundary) (Tubes 3,6,11,22,27,47,58)
- 5 are fully modeled as 360° (Tubes 15,31,39,51,63)

See Figure 3 for the locations of the penetrations modeled. The dimensions were provided in Reference 3.

2.2.1 Hemispherical Head CRDM Penetration Dimensions

Based on Reference 3, the penetration hole in the hemispherical head is 4.0 inches in diameter. The resulting interference fit region begins at the outside surface of the hemispherical head and extends down to the toe of the J-groove weld (at the top of the weld butter - see Figure 4a).

For this analysis, the interference fit region was modeled as beginning at the top edge of the weld butter CRDM to hemispherical head weld and extending to the outer surface of the head. The weld was not specifically modeled. This resulted in layout as shown in Figure 4b.

2.2.2 CRDM Tube Dimensions

Per Reference 4, the CRDM tube outside diameter at the penetration is 4.025 inches with an inner diameter at 2.765 inches.

For this analysis, the tube outside diameter is set at a constant 3.998 inches and inner diameter at 2.738 inches (to maintain the original 0.63 inch wall thickness). The outside CRDM diameter of 3.998 inches allows for a 0.001 inch radial gap between the CRDM tube and the hemispherical head hole (modeled at 4.00 inches diameter). This gap was necessary to support CONTAC52 elements, which were used to simulate the interference fit between the CRDM and the hemispherical head penetration holes (see Section 5.6 for additional details on the interference loading).

The effects of the modeled reduction in outside diameter (and corresponding inside diameter) were considered insignificant for this gap evaluation. In addition, due to the variation in interference values (Reference 5) between each tube, any variations from the drawing dimensions from References 3 and 4 were further minimized.

In addition, neither the tube expansion nor the bolted flange connections (both of which occur outside the reactor vessel) are modeled and the total height of the tube beyond the hemispherical head is arbitrary.

Finally, the actual CRDM tubes are intended to project through and slightly into the hemispherical head. This projection was not modeled.

2.2.3 CRDM-to-Head Weld

A key set of dimensions was the varying height of the CRDM tube to hemispherical head weld. The weld height varies around the circumference of the tube based on each tube's position on the hemispherical head (see Figure 4a and Reference 3).

While the specific weld and weld material were not modeled, the weld attachment height is important and must be included. To determine these heights, the following observations and assumptions were made:

- The $^3/_{16}$ inch butter (though not modeled specifically) was not considered as part of the weld attachment height calculation. The butter was not modeled but considered part of the hemispherical head.
- The vertical distance from the top edge of the weld (bottom of the butter) to the corner of the base metal of the hemispherical head is a constant $^{25}/_{32}$ inches anywhere on the weld and for every tube (Reference 3).
- The diameter between the opposite centers of curvature for all of the CRDM tube to hemispherical head welds is a constant $4^1/_4$ inches.
- The weld height varies linearly around the circumference [3].

Based on Reference 6, the radial extent of the J-groove weld (including the butter) from the radius of curvature to the outside surface of the base metal is 0.8606 inches. The resulting edge-to-edge diameter of the weld prep is therefore 4.25 inches + 2 * 0.8606 inches or 5.9712 inches.

A simple ANSYS model was thus developed that penetrates the inner surface of the hemispherical head at each tube location using the diameter of 5.9712 inches. For this model it is necessary to specifically exclude the inner clad, resulting in an inner surface radius for the hemispherical head of $87^1/_4$ inches [2]. The resulting intersections of the 5.9712 inch diameter penetration and the inner hemispherical head surface were then shifted $^{25}/_{32}$ inches up to determine the top height of the welds. The ANSYS file used for this study is named WELD.INP and included with the Project CD-ROM. The resulting heights are also included in Appendix A.

In the final finite element model the height values determined above were used in conjunction with the actual modeled CRDM tubes to create a series of angled planes, which were used to divide the CRDM tube at the top of the weld (see Figure 4b).

The final weld connection between the hemispherical head and the CRDM tubes is via a series of degree-of-freedom couples between the nodes along the inner surface of the hole in the hemispherical head and the outer surface nodes of the CRDM tubes. These couples are only applied along the modeled weld height and can be seen in Figure 5.

3.0 Materials

Reference 2 indicates the following materials were used for the modeled components. Note that the closure stud was not actually modeled and its properties are not included in this evaluation.

Component	Material
Upper Head	SA-533 Grade B Class 1
Closure Flange	SA-508 Class 2
CRDM Housing Tube	SB-167 (Alloy 600)
Closure Stud	SA-540 B23 Class 3

No welds were specifically modeled nor were the weld materials included. Where material changes occur across welds, the material was simply modeled as an instant change. In the case of the hemispherical head to closure flange, the material is assumed to change at the end of the fillet region farthest from the closure flange.

The material properties used for this evaluation are based on the 1989 ASME Code [7] for a temperature of 600°F. The properties used are indicated in the following table:

Material	Modulus of Elasticity E, psi	Mean Coefficient of Thermal Expansion, α, in/in/°F
SA-533 Grade B Class 1	26.4e6	7.83e-6
SA-508 Class 2	26.4e6	7.42e-6
SB-167 (Alloy 600)	28.7e6	7.82e-6

A Poisson's Ratio of 0.3 was used for all of the materials, as was the metal density of 0.283 lb/in³.

4.0 Mechanical Boundary Conditions

Symmetry boundary conditions were applied along the circumferential free ends of the model. See Figure 6 for these boundaries.

In the case of the closure surface, a series of CONTAC52 gap elements were developed at the compression surface (see Figure 2). The gap elements attach the compression surface of the closure flange to a series of nodes that are fixed in the vertical direction. These nodes simulate the compression surface of the lower (un-modeled) flange. The two nodes that make up this contact region are 0.1 inches below the compression surface of the modeled flange, but behave as if they are in direct contact. In addition, the node pairs of each gap element are coupled for horizontal translations and have a weak spring element (COMBIN14, k=100 lb/in) between them. These additions are included to provide initial numeric stability in the analysis. See Figure 7 for applied couples and vertical restraint on gap elements.

5.0 Loading

Two gap evaluations were performed under normal operating conditions. The only variation between the two evaluations was the interference loads between the CRDM tube and the hemispherical head. The loads that exist for the normal operating condition are defined in the following sections as are interference loads used in the two gap evaluations.

5.1 Temperature

A uniform temperature of 605°F [8] was applied over the entire model with the stress free temperature being 70°F.

5.2 Pressure

The normal operating pressure is 2155 psi for Davis-Besse, Unit 1 [8]. The pressure was applied to the inside surface of the hemispherical head, the hemispherical head side end of the CRDM tube, and to the flange closure face out to a radius of 84.8115 inches [2].

In addition, a cap pressure was applied to the outside free end of the CRDM tubes to simulate line load in each tube. The pressure was calculated as:

$$P_{cap\_tube} = \frac{P \cdot r_{inside}^2}{\left(r_{outside}^2 - r_{inside}^2\right)} = \frac{2155 \cdot 1.37^2}{\left(2.0^2 - 1.37^2\right)} = 1905.1 \text{ psi}$$

Note that the applied cap load was actually applied in the negative direction in ANSYS, thus providing a traction load. See Figure 8 for the applied pressure surfaces.

5.3 Closure Bolt Load

A total closure bolt load of 84.0e6 lbs is specified in Reference 8. As there are a total of 60 bolts, the total load per bolt is 1.4e6 lbs. The closure bolt load was simulated by applying a pressure load at the top of the flange on each of the bolt hole locations that were modeled. The area of each hole is $\pi*3.5^2$ or 38.48 in^2; thus the applied pressure is 36378.27 psi. See Figure 9 for the applied pressures for the bolt load simulation. Using pressure allows a more rapid model development and should involve no significant loss of accuracy since the areas of interest were the CRDM tubes. In addition, the overall size of the closure flange relative to the rest of the head minimizes any stiffness changes had the holes been modeled and the actual studs included.

5.4 Gasket/Spring Loads

During closure, there are three other active loads applied to the upper flange; two gasket loads and a spring load. The gasket squash loads and their radius of application were defined in References 8 and 2, respectively, as:

- Inner Gasket: Total Squash Load = 400,000 lb. at a radius of 84.8115 inches
- Outer Gasket: Total Squash Load = 407,600 lb. at a radius of 86.4365 inches

The gasket loads are applied as a series of nodal loads at the bottom of the flange in a positive vertical direction. The total squash load for a 45° section of the model is 50,000 lbs and 50,950 lbs for the inner and outer gaskets, respectively. At each radius of load application there are 37 equally spaced circumferential nodes (for a total of 74 nodes in two lines, 37 for the inner gasket and 37 for the outer gasket with the two rows of nodes lying side by side in the finite element model).

The total load on the inner 35 nodes for each of the gaskets was 1,388.89 lbs for the inner and 1,415.28 lbs for the outer gasket. The 2 symmetry edge nodes for the inner gasket are loaded with 694.45 lbs while the outer gasket symmetry nodes received 707.64 lbs. Figure 10 shows the applied load as two sets of small upward arrows nearest the outside edge of the closure flange.

The spring load (also referred to as "ledge" load) is the reaction of the plenum cover and core support shield assembly (not modeled for this evaluation) to the applied closure stud preload. For the upper closure flange the response load and its radius of application was defined in Reference 8 as:

- Spring Load: Total Load = 6.0e6 lb. at an assumed radius of 80.5 inches

The spring load was simulated in the same manner as the gasket loadings; it was simulated with a series of evenly spaced circumferential nodal loads. The total load for the 45° model was 750,000 lbs. The inner 35 nodes therefore received a load of 20,833.33 lbs in the positive vertical direction while the symmetry edge nodes were loaded at 10,416.67 lbs. Figure 10 shows the applied load as a row of large upward arrows furthest from the outside edge of the closure flange.

5.5 Deadweight Load

A 1-g positive vertical acceleration (in ANSYS a positive acceleration produces the desired deadweight load) was applied to the model to simulate gravity for the deadweight load. Although the entire tubes are not modeled nor are any other extraneous components, the total effect is expected to be minimal in comparison to the total weight of the head.

5.6 CRDM Housing Interference Load

The final load applied to the finite element model was the interference loads between the tube outside surfaces and the inside interference zone of the hemispherical head. This load was the only load that changed between the two gap analyses. The interference dimensions at the top and the bottom of each tube at Davis-Besse, Unit 1 was provided in Reference 5.

The interference values for each tube will be varied linearly from the top edge of the hemispherical head down to the bottom of the interference zone (just above the weld). Note that because of the layout of the interference zone the interference values will vary around the circumference of the tube for a given height. This is due to the changing interference zone height of each tube as a result of the slope of the hemispherical head (see Figure 4)

The first gap analysis will be used to support a leak rate calculation based on the gap openings between the CRDM and the hemispherical head. As such the worst (or largest) interference values were modeled to minimize gap opening and thus leak rate. In addition, with only 13 of the 69 tubes modeled, the worst interference load for the corresponding tube sets was used. Worst case for this analysis was the tube that had the greatest top or bottom interference dimension.

Table 1 lists the modeled tube numbers, the tube numbers in the corresponding tube set, the worst case tube, and the resulting interference dimensions for that tube. All of the tube interference values are included in Appendix B. The final ANSYS input file for the gap evaluation was named DBCRDM.INP and is included on the project CD-Rom.

Table 1

Interference Values for Gap Evaluation Supporting Leak Rate Evaluation

Modeled Tube #	Corresponding Tube #	Worst Tube #	Diametrical Interference (Gap) Dimensions (in)	
			Top	Bottom
1	-	1	0.0001	0.0012
3	2,4,5	2	0.0019	0.0020
6	7,8,9	6	0.0012	0.0006
11	10,12,13	10	0.0013	0.0002
15	14,16,17,18,19,20,21	17	0.0014	0.0013
22	23,24,25	24	0.0016	0.0004 (Gap)
27	26,28,29	27	0.0011	0.0005
31	30,32,33,34,35,36,37	33	0.0018	0.0003
39	38,40,41,42,43,44,45	45	0.0014	0.0011
47	46,48,49	48	0.0009	0.0013
51	50,52,53,54,55,56,57	50	0.0021	0.0010
58	59,60,61	61	0.0012	0.0003
63	62,64,65,66,67,68,69	63	0.0014	0.0015

The second gap analysis will be used to support a fracture mechanics evaluation of the CRDM weld cracking. To support this evaluation the least (or smallest) interference values were modeled to maximize gap opening and thus maximize crack opening in the follow-on flaw evaluation. Again, with only 13 of the 69 tubes modeled, the least interference load for the corresponding tube sets was used. Least for this analysis was the tube that had the smallest top or bottom interference dimension. Note that in a number of cases there was no interference but an actual gap instead. Any actual gaps were included in the model.

Table 2 lists the modeled tube numbers, the tube numbers in the corresponding tube set, the best case tube, and the resulting interference dimensions (or gap dimensions) for that tube. All of the tube interference values are included in Appendix B. The final ANSYS input file for the gap evaluation was named DBCRDM-O.INP and is included on the project CD-Rom.

Table 2

Interference Values for Gap Evaluation Supporting Flaw Evaluation

Modeled Tube #	Corresponding Tube #	Best Tube #	Diametrical Interference (Gap) Dimensions (in)	
			Top	Bottom
1	-	1	0.0001	0.0012
3	2,4,5	5	0.0007	0.0009
6	7,8,9	7	0.0001	0.0009
11	10,12,13	13	0.0009	0.0001 Gap
15	14,16,17,18,19,20,21	14	0.0004 Gap	0.0005 Gap
22	23,24,25	24	0.0016	0.0004 Gap
27	26,28,29	28	0.0004	0.0008
31	30,32,33,34,35,36,37	35	0.0002	0.0010 Gap
39	38,40,41,42,43,44,45	44	0.0012	0.0002
47	46,48,49	49	0.0002	0.0002 Gap
51	50,52,53,54,55,56,57	54	0.0000	0.0001 Gap
58	59,60,61	59	0.0008	0.0001
63	62,64,65,66,67,68,69	67*	0.0005	0.0006

*Tube 67 was selected in lieu of Tube 65 (Top Interference = 0.0010 inches, Bottom Interference = 0.0004), as the average interference of Tube 67 from top to bottom was lower than Tube 65.

For both evaluations, the application of the interference (or gap) was via a CONTAC52 gap element. The CONTAC52 element allows the entry of a negative gap value, which is treated as an interference value rather than the typical positive physical gap. Each tube was thus modeled with a series of gap elements simulating the specific interference value. The interference values entered into ANSYS were halved, as the ANSYS element was established as a radial gap. The values were also evenly spaced down the interference zone and varied linearly from the top interference value to the bottom, resulting in a total of 5 sets of interference values around the circumference of the tube for each modeled tube. The use of CONTAC52 elements in this application was verified in a separate study shown in Appendix C.

6.0 Results

For each evaluation, a series of gap results were determined via a post-processing file named POST. The post-processing file captured the element number, the gap opening and the gap's position relative to its specific tube. Specific results for each evaluation are detailed in the following sections.

6.1 Gap Opening Evaluation – Leakage Evaluation

For this evaluation, where the greatest CRDM interference values were used, it was determined that tubes 1 and 3 do not have a vertical path that would allow leakage (see Figure 11). In both cases, the blockage occurs at the bottom of the interference zone (i.e. just above the weld). All of the other tubes have a vertical path where leakage is possible. Table 3 lists each specific tube's smallest gap opening along a vertical path anywhere in the interference zone whose gaps are all open. For tubes 1 and 3 the minimum interference value is listed in the form of a negative value. A complete list of tube results can be found in the Excel spreadsheet DB-GAP.XLS (included on the project CD-Rom).

Table 3
Minimum Gap Results for Leakage Evaluation

Tube Number	Minimum Gap (inches)
1	-0.00000367 (Interference)
3	-0.00002483 (Interference)
6	0.000073863
11	0.000012591
15	0.000011731
22	0.000019860
27	0.000081417
31	0.000066524
39	0.000020384
47	0.000082758
51	0.000000682
58	0.000102970
63	0.000001171

As a result of Tube 3's lack of a gap condition, further investigation of the other tubes on the group of Tubes 2, 3, 4 and 5 were evaluated. The results of these evaluations are included in Appendix D of this calculation package.

6.2 Gap Opening Evaluation – Fracture Mechanics Evaluation

For this evaluation, where the least CRDM interference values were used (in some cases actual physical gaps were used), a series of gap values were determined to support future fracture mechanics evaluations. Specifically, gap information axially along the tube's interference zone at the highest point of the weld (uphill side) and gap information axially along the tube's interference zone at the lowest point of the weld (downhill side) were determined.

Table 4 lists each specific tube's gap opening as it varies along a vertical path in the interference zone at the uphill and downhill sides. A complete list of tube results can be found in the Excel spreadsheet, DB-BIG.XLS (included on the project CD-Rom).

Table 4
Tube Gap Results for Fracture Mechanics Evaluation

Tube #	Hill Side	Gap (inches) at Interference Zone Level						
		Top	Top/Mid #1	Top/Mid #2	Middle	Mid/Bot #1	Mid/Bot #2	Bottom
1	-	0.0017027	0.0016565	0.0015937	0.0015723	0.0015353	0.0011360	-0.0000020
3	Uphill	0.0024574	0.0023365	0.0021809	0.0020339	0.0018463	0.0013500	0.0001556
	Downhill	0.0003352	0.0005402	0.0007489	0.0010388	0.0013316	0.0011618	0.0001205
6	Uphill	0.0028988	0.0026593	0.0023999	0.0021418	0.0018442	0.0012589	0.0000598
	Downhill	-0.0000026	0.0001903	0.0004334	0.0007807	0.0011477	0.0010052	0.0000002
11	Uphill	0.0014143	0.0015783	0.0017222	0.0018428	0.0018881	0.0016480	0.0006723
	Downhill	0.0008567	0.0010146	0.0012106	0.0015580	0.0019643	0.0017880	0.0006789
15	Uphill	0.0036170	0.0034250	0.0031924	0.0029331	0.0026183	0.0020375	0.0008296
	Downhill	-0.0000355	0.0001316	0.0005409	0.0011180	0.0017579	0.0017561	0.0007555
22	Uphill	0.0019850	0.0021062	0.0021694	0.0021753	0.0021219	0.0018306	0.0008272
	Downhill	-0.0000361	-0.0000139	0.0002112	0.0008535	0.0017436	0.0019320	0.0008897
27	Uphill	0.0016772	0.0017365	0.0017705	0.0017703	0.0016893	0.0013648	0.0002833
	Downhill	0.0009714	0.0009983	0.0011108	0.0014537	0.0019235	0.0016472	0.0002996
31	Uphill	0.0033857	0.0033432	0.0032294	0.0030558	0.0028078	0.0023198	0.0011671
	Downhill	-0.0000648	-0.0000090	0.0004283	0.0011914	0.0020943	0.0021780	0.0011045
39	Uphill	0.0025787	0.0025887	0.0025279	0.0023936	0.0021773	0.0017300	0.0006024
	Downhill	-0.0000846	-0.0000535	0.0000485	0.0007372	0.0016514	0.0016877	0.0005377
47	Uphill	0.0018693	0.0020580	0.0022058	0.0022682	0.0022081	0.0019444	0.0008877
	Downhill	0.0011666	0.0011483	0.0013298	0.0018660	0.0025989	0.0023482	0.0009046
51	Uphill	0.0039365	0.0038082	0.0018660	0.0032373	0.0028077	0.0021742	0.0008409
	Downhill	-0.0000887	-0.0000450	0.0002246	0.0010571	0.0020923	0.0020855	0.0007673
58	Uphill	0.0031386	0.0030919	0.0029563	0.0027254	0.0024084	0.0019036	0.0007196
	Downhill	-0.0000573	-0.0000342	0.0000799	0.0009045	0.0019880	0.0020242	0.0007205
63	Uphill	0.0034363	0.0033127	0.0030870	0.0027611	0.0023475	0.0017659	0.0005004
	Downhill	-0.0001121	-0.0000789	-0.0000041	0.0008004	0.0018993	0.0018984	0.0004711

7.0 References

1) ANSYS Mechanical, Revision 5.7, ANSYS Inc., December 2000

2) Framatome Technologies Technical Document 33-1201205-02, "Stress Report Summary for Reactor Vessel, Toledo Edison Company, Davis-Besse Unit No. 1," SI File W-ENTP-11Q-219P

3) B&W Drawing 154628E, Rev. 5, "Closure Head Sub-Assembly," Davis-Besse Document Number M-503-212, SI File No. W-ENTP-11Q-219P

4) Email from Prasoon Goyal (First Energy) to Richard Bax (SI), dated Thursday, September 20, 2001, 4:45 AM, "Additional Design Input," Referencing B&W Drawing 154632E, Rev. 2, "Control Rod Mechanism Housing," Davis-Besse Document Number M-503-213-2, SI File No. W-ENTP-11Q-219P

5) Framatome Document 51-5013435-02, "CRDM Nozzle/Bore Dimensional Analysis," SI File No. W-ENTP-11Q-219P

6) Structural Integrity Calculation W-ENTP-07Q-301, Rev. 0, "CRDM Penetration J-Weld Size Calculation"

7) ASME Boiler and Pressure Vessel Code, 1989 Edition, Section III, Appendices

8) Letter DBE-01-000133, Dated September 13, 2001 from Prasoon Goyal (First Energy) to Dick Mattson (SI), SI File W-ENTP-11Q-219P

9) Email from Prasoon Goyal (First Energy) to Richard Bax (SI), dated Wensday, September 19, 2001, 12:39 PM, "Additional Design Input," Referencing B&W Drawing 54628E "Closure Head Sub-Assembly," SI File No. W-ENTP-11Q-219P

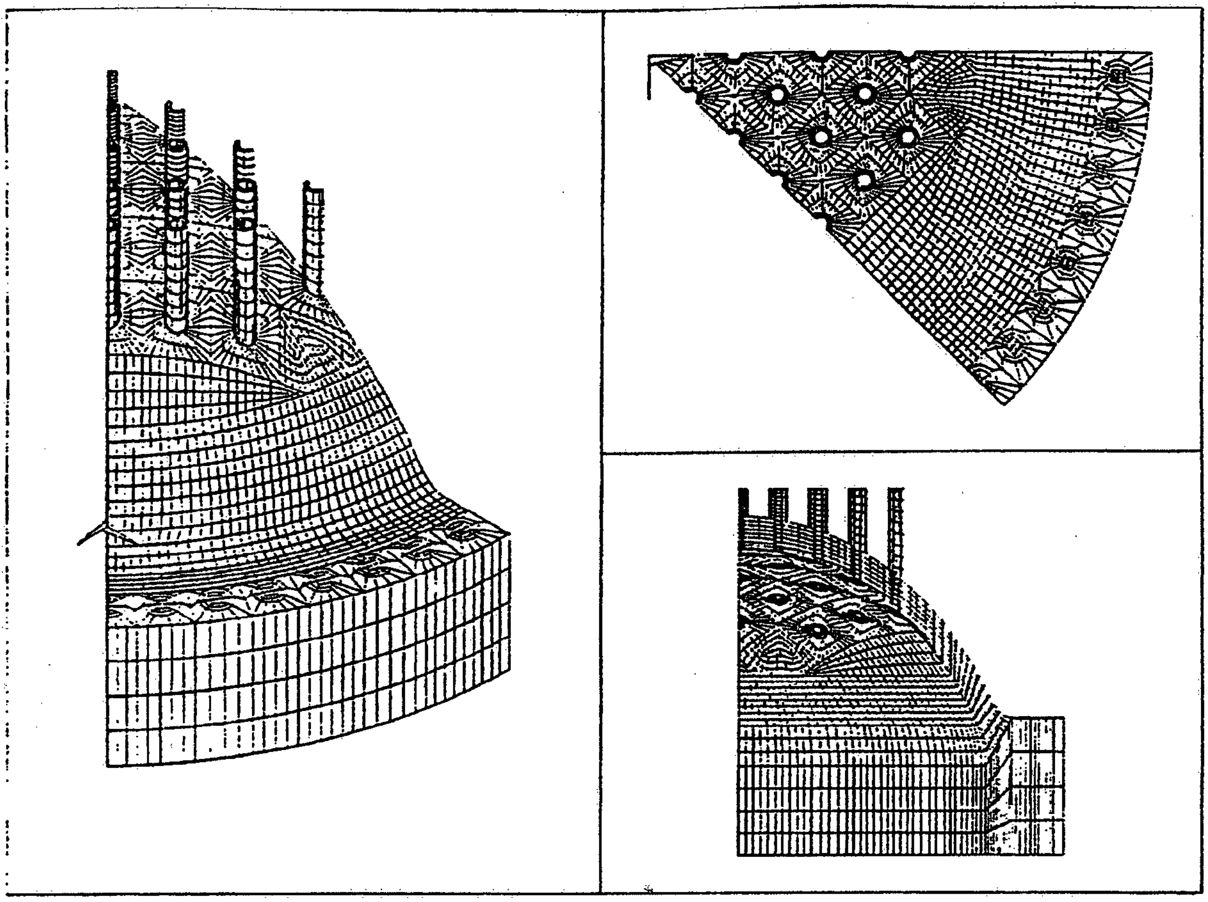

Figure 1 – Finite Element Model

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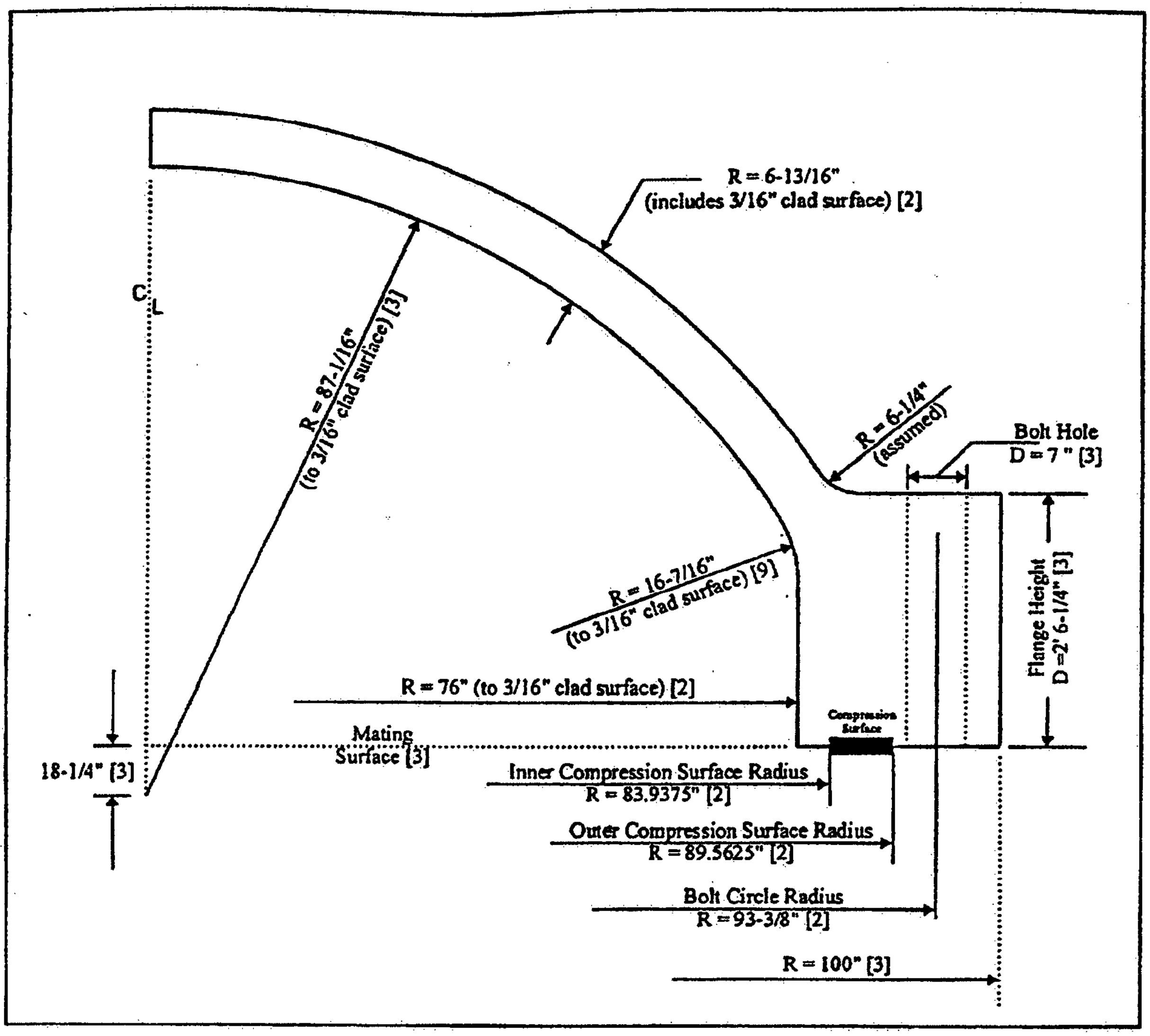

Figure 2 – Top Hemispherical Head / Closure Flange Dimensions

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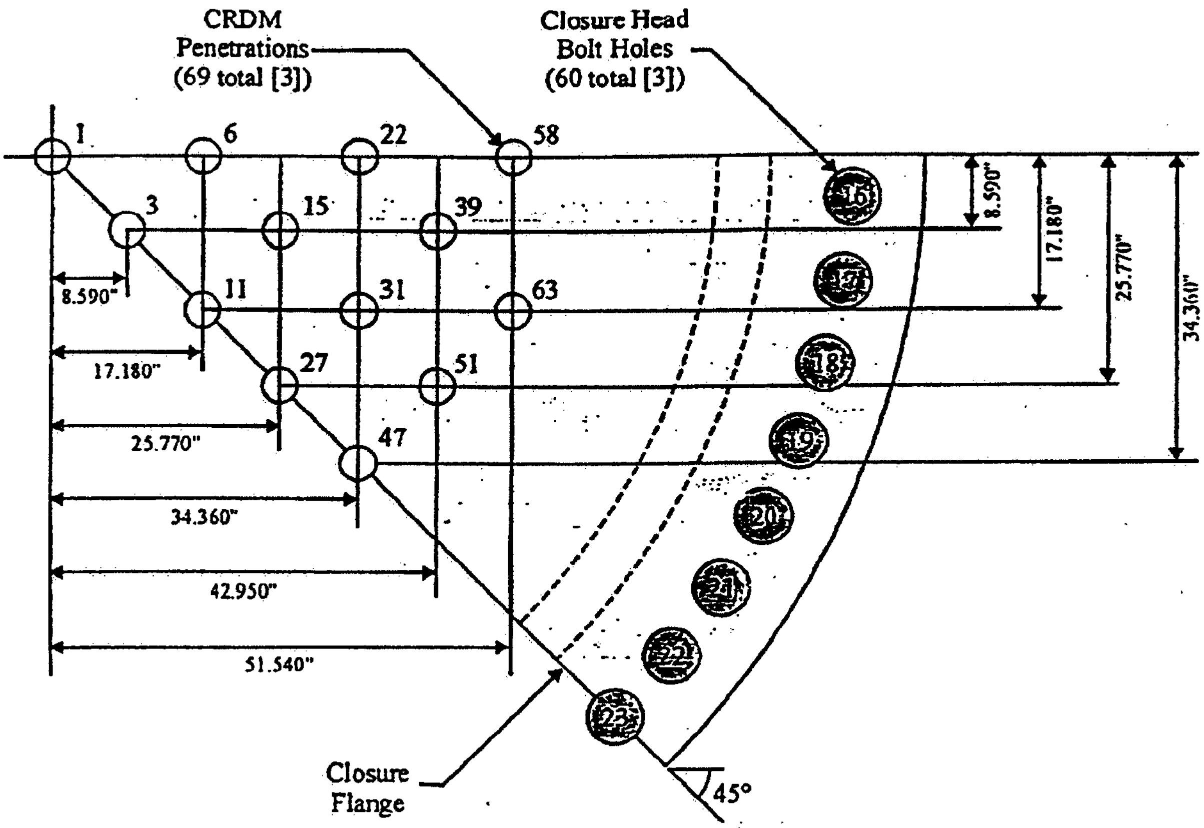

Figure 3 – CRDM Penetration Locations

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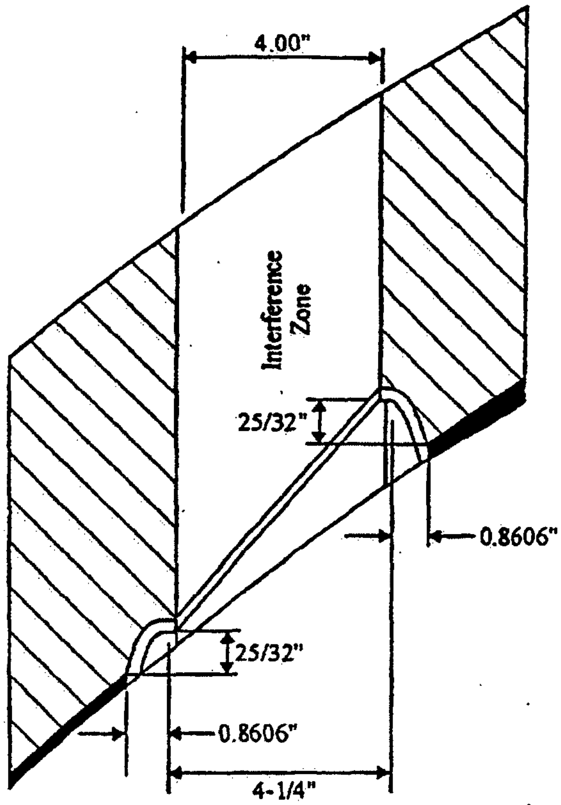

Figure 4a
From Reference 3

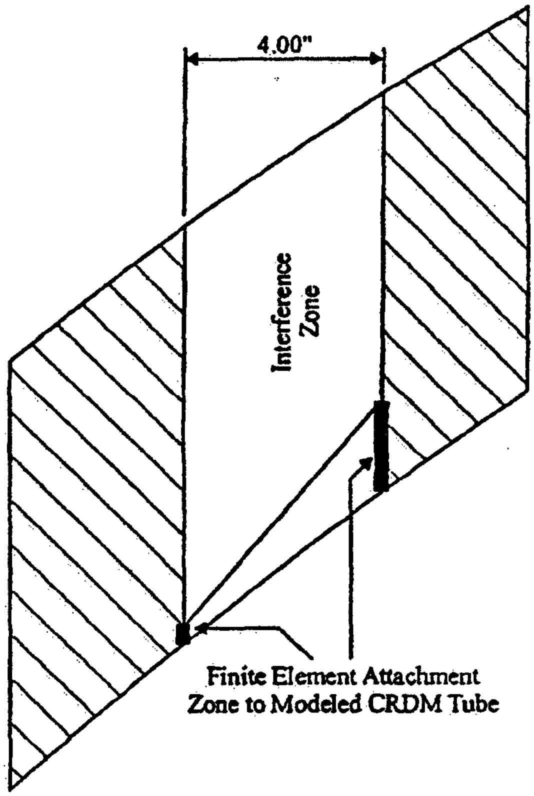

Figure 4b
As-Modeled

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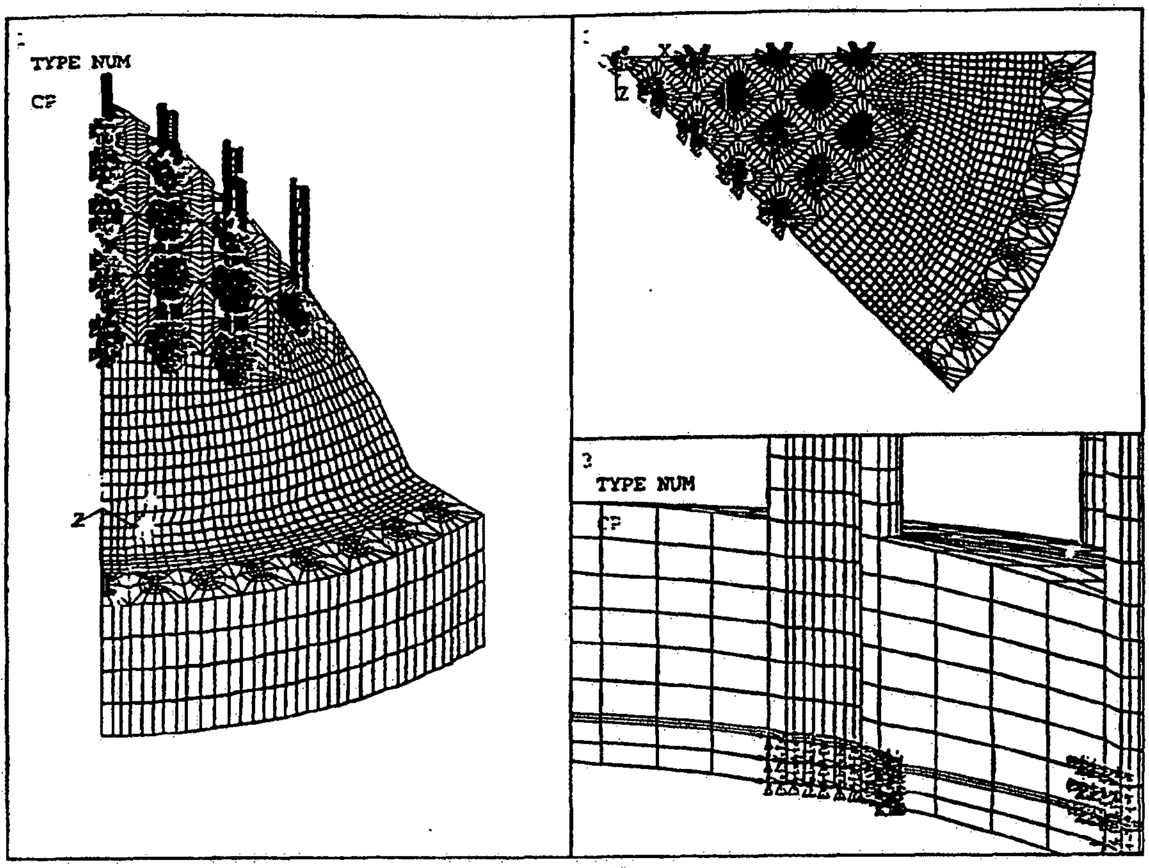

Figure 5 – Applied Couples Attaching CRDM Tube to Hemispherical Head

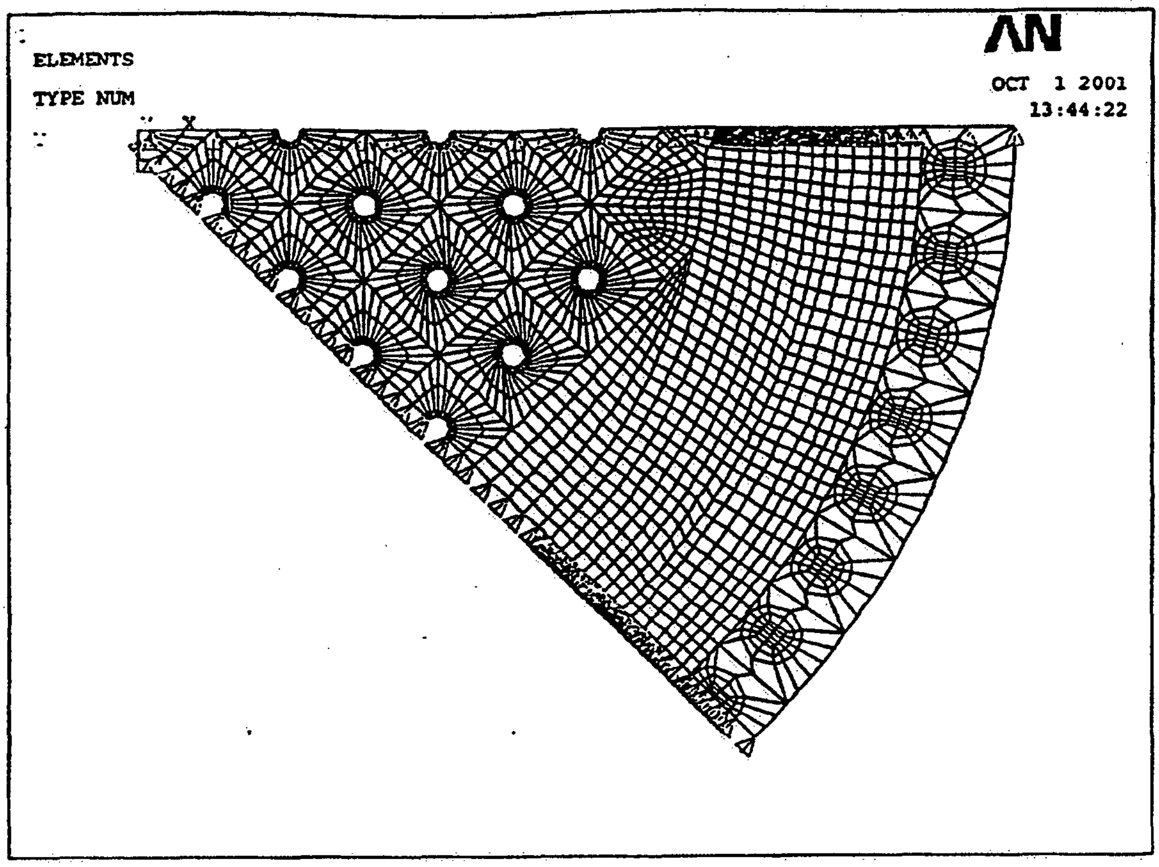

Figure 6 – Applied Symmetry Boundary Conditions

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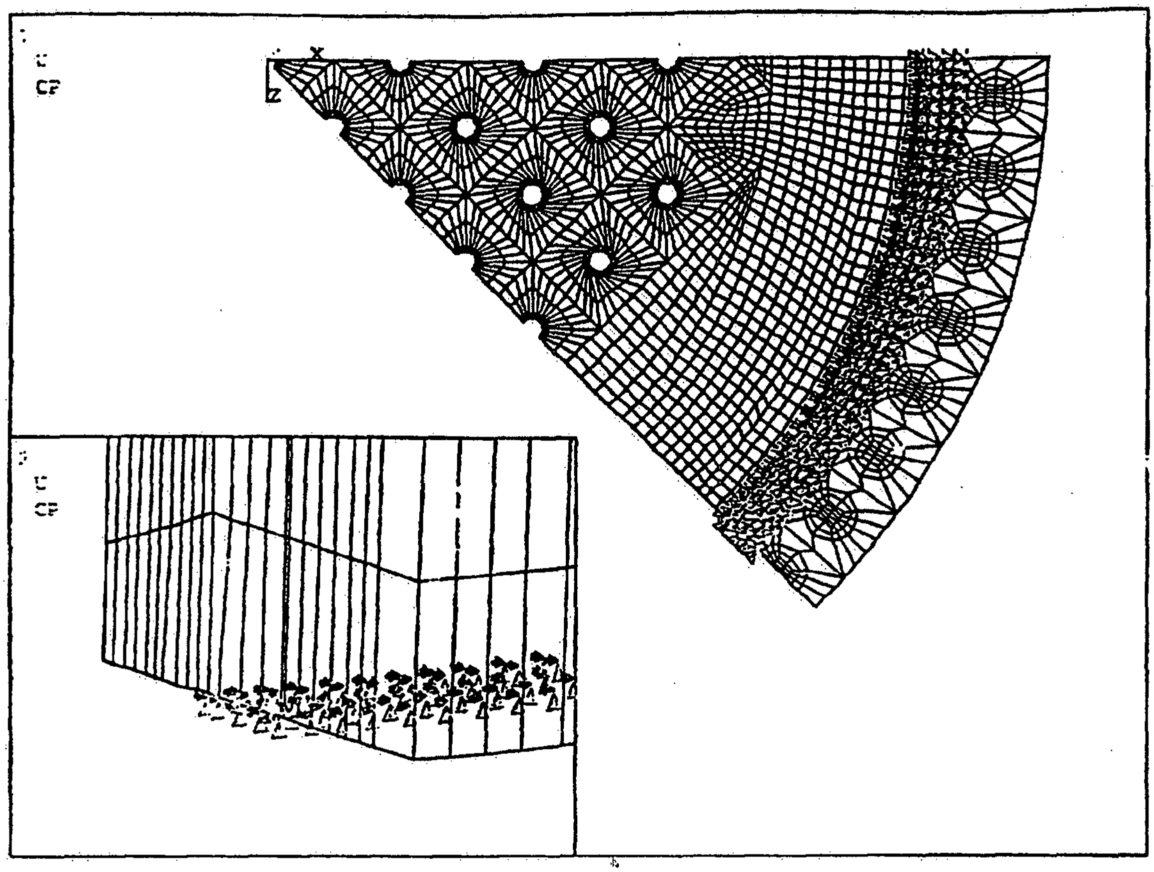

Figure 7 – Applied Couples and Vertical Restraint at Contact Surface
(Lower Flange Contact Simulated with Gap Elements)

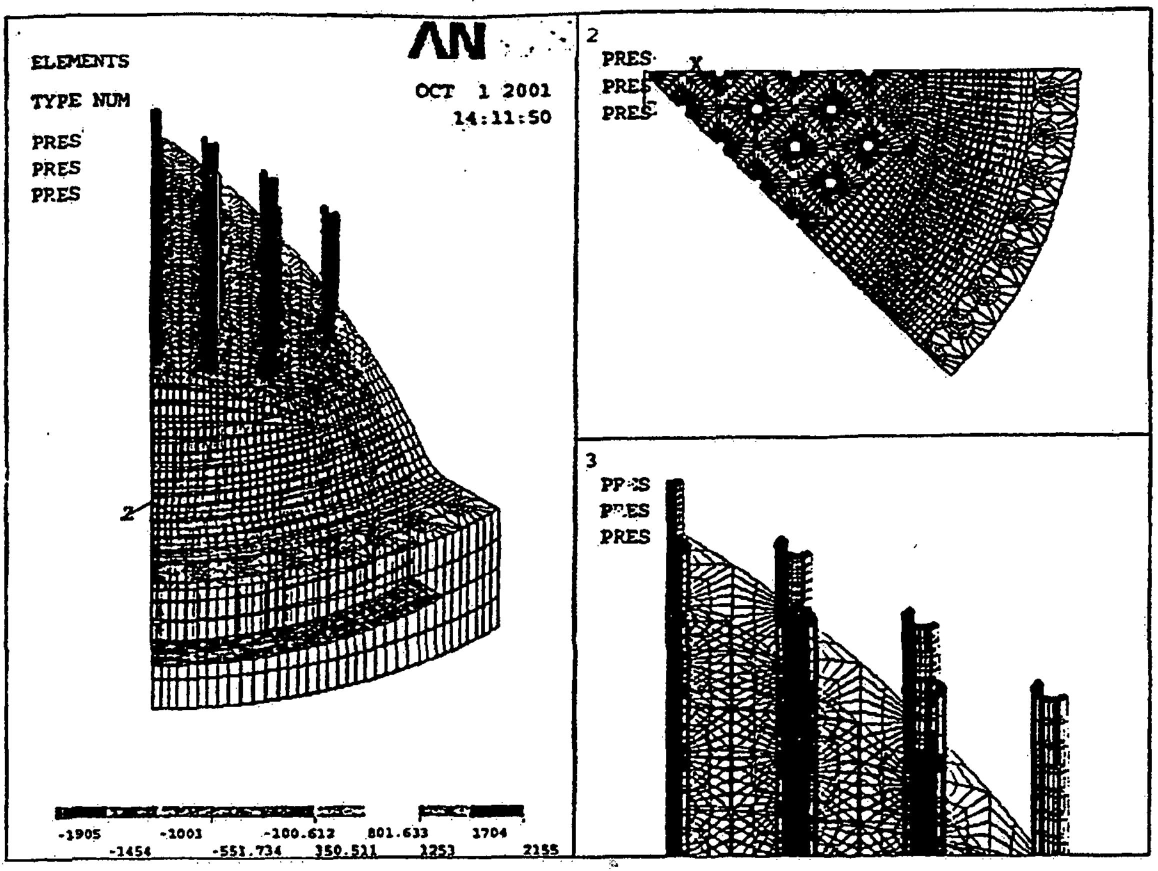

Figure 8 – Applied Normal Operating Pressure Load

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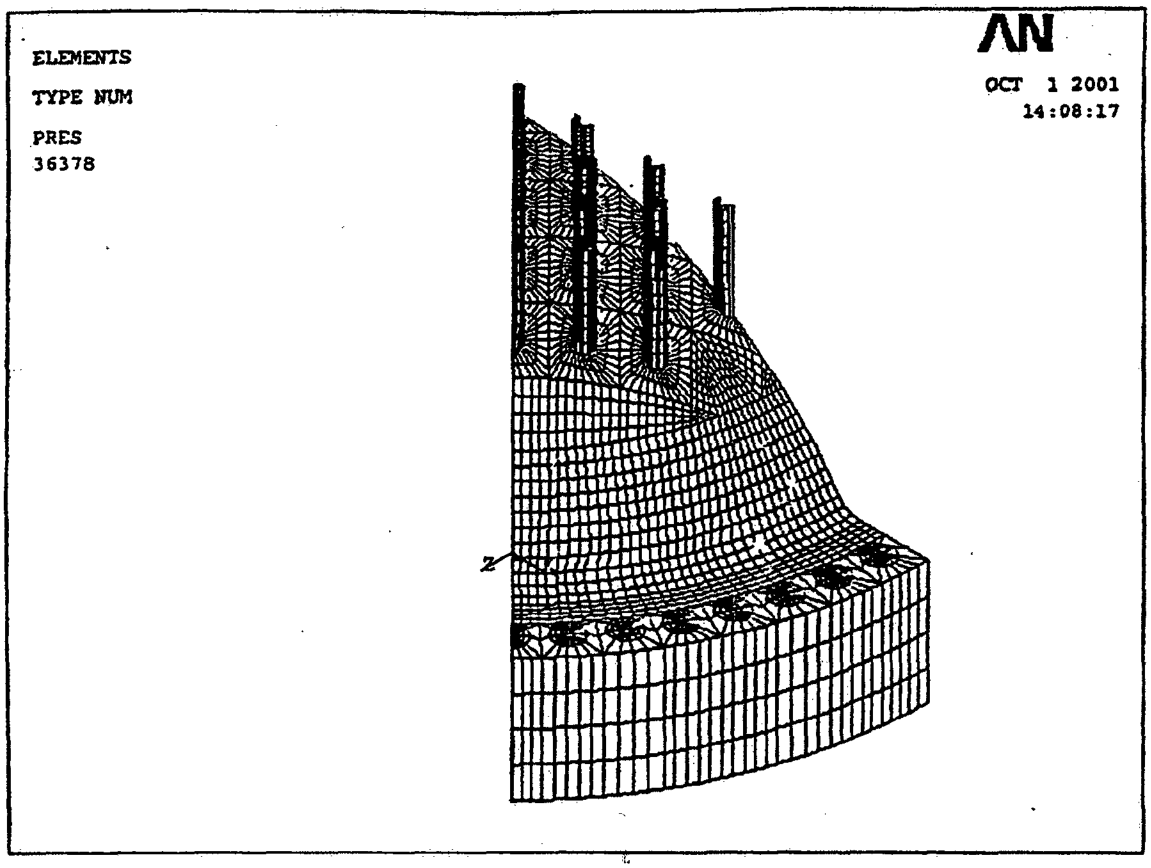

Figure 9 – Applied Pressures To Simulate Closure Bolt Load

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Framatome ANP Proprietary

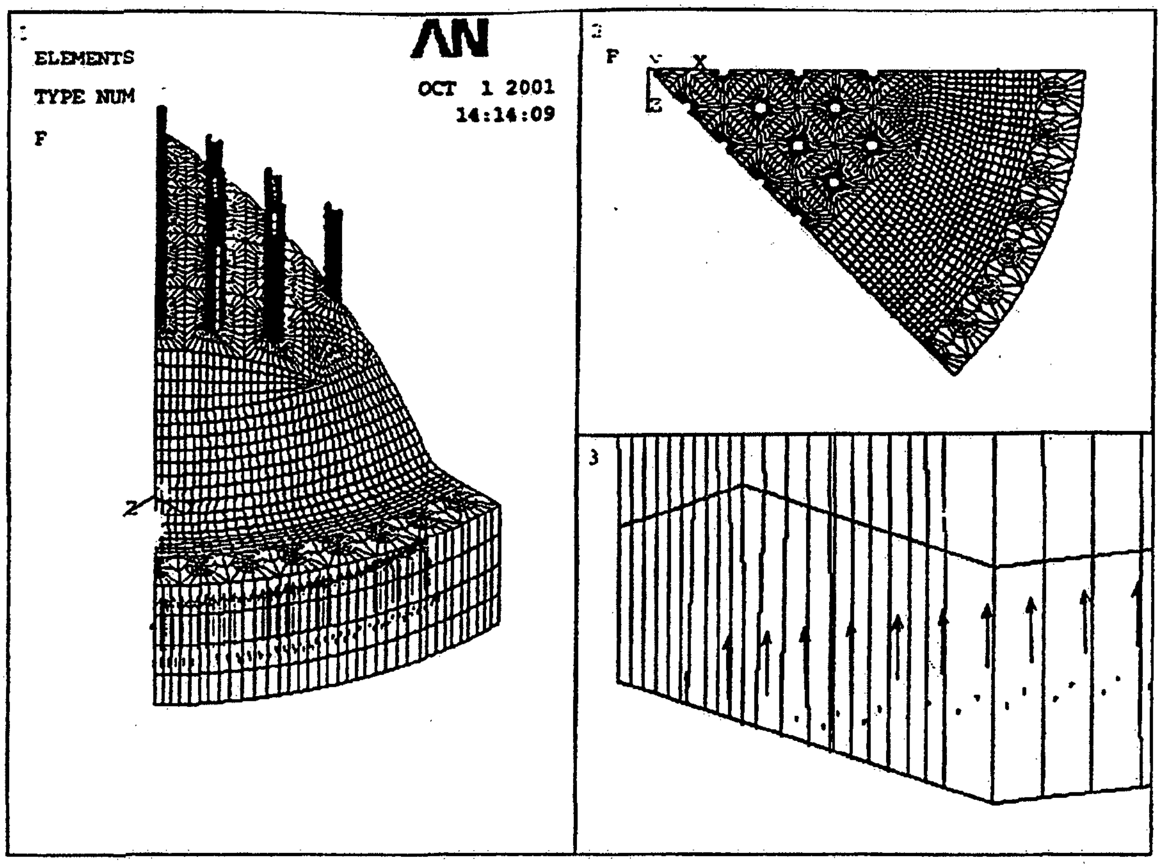

Figure 10 – Applied Gasket and Spring Loads

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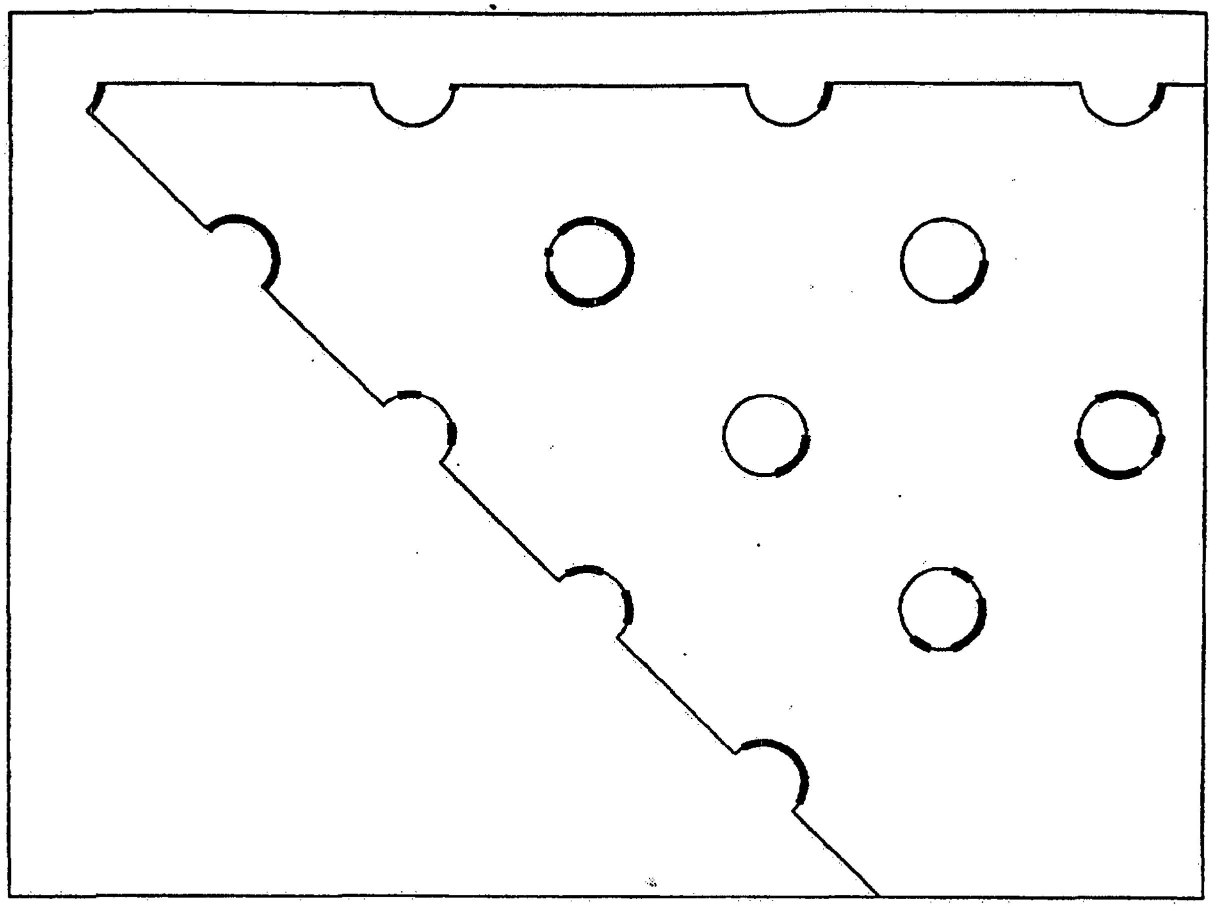

Figure 11 – Locations Where Gaps Are Closed Anywhere Along the Vertical Path of the Interference Zone (Blue Lines) for Worst Case Interference Values (Leakage Evaluation)

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APPENDIX A

CRDM to Hemispherical Head Weld Heights Resulting From ANSYS Input File WELD.INP

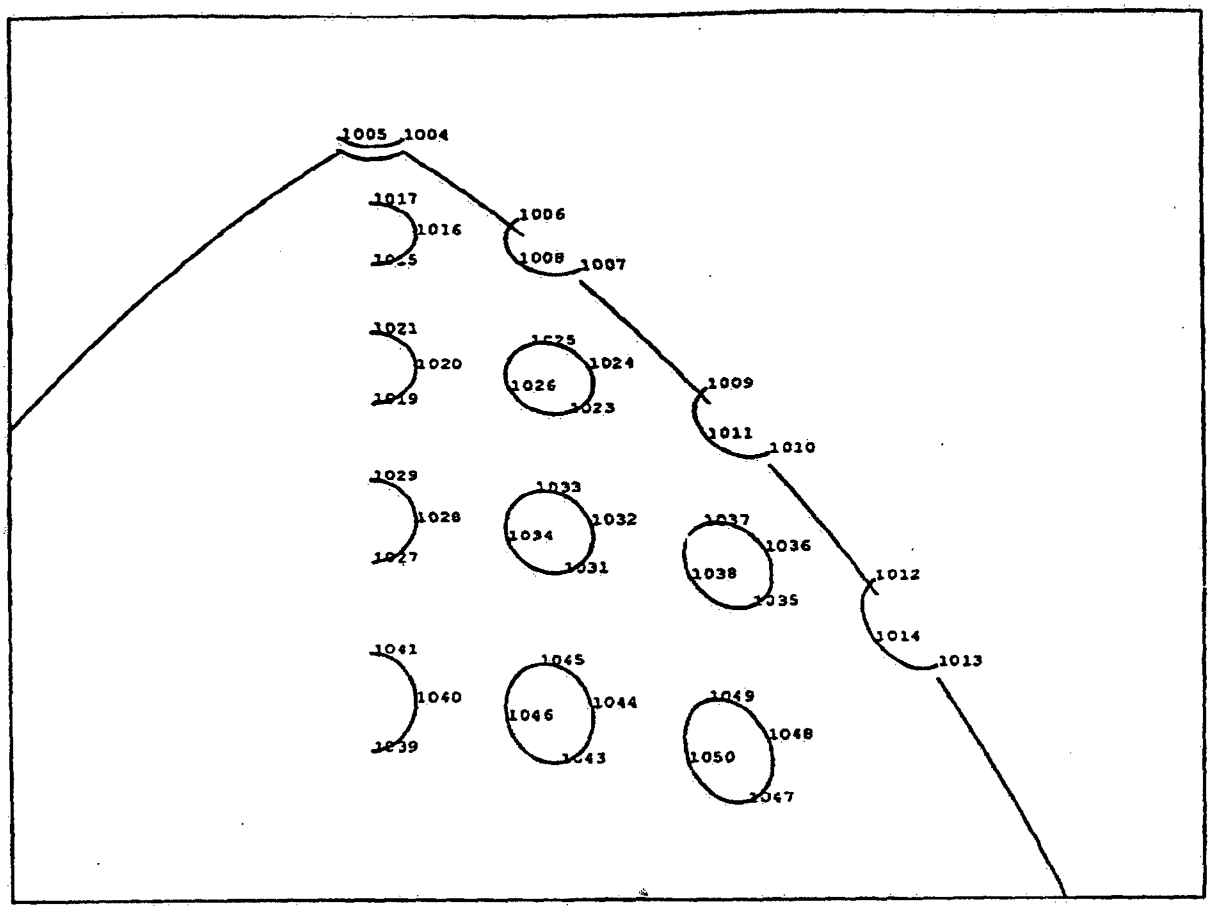

Figure A-1 Key Point Numbers Of Calculated Weld Heights

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The following table lists the height of the interface between the hemispherical head inside surface (excluding the clad) and overall weld diameter for the tubes (5.9712 inches from Section 2.2.3). The height values are for the point on the weld circle nearest the top dead center of the hemispherical head and a height at the farthest.

Also included in the table is height at the top of the weld, which includes the $^3/_{16}$ inch weld butter. The actual finite element model excludes the butter as part of the weld so the "Top of Weld Value" minus the $^3/_{16}$ inches was used in the final finite element model to develop a plane that divided the modeled CRDM tube structure along the top of the weld. All of these values were developed in a preliminary ANSYS input file named WELD.INP (include on the project CD-Rom).

Modeled Tube #	Location From Top-Dead Center	Height Measured from Center of Curvature of Hemispherical Head			
		Node #	Inside Surface Height (in)	Node #	Top of Weld Height (in)
1	Nearest	4	87.1989	1004	88.2614
3	Nearest	6	86.08764	1006	87.15014
	Furthest	7	84.88764	1007	85.95014
6	Nearest	9	81.02136	1009	82.08386
	Furthest	10	78.37205	1010	79.43455
11	Nearest	12	72.4916	1012	73.5541
	Furthest	13	68.11403	1013	69.17653
15	Nearest	17	86.76757	1017	87.83007
	Furthest	15	85.92749	1015	86.98999
22	Nearest	21	84.60745	1021	85.66995
	Furthest	19	82.87501	1019	83.93751
27	Nearest	25	83.83298	1025	84.89548
	Furthest	23	81.87531	1023	82.93781
31	Nearest	29	80.57964	1029	81.64214
	Furthest	27	77.83217	1027	78.89467
39	Nearest	33	79.73252	1033	80.79502
	Furthest	31	76.80169	1031	77.86419
47	Nearest	37	77.11485	1037	78.17735
	Furthest	35	73.6452	1035	74.7077
51	Nearest	41	74.38134	1041	75.44384
	Furthest	39	70.3724	1039	71.4349
58	Nearest	45	73.44343	1045	74.50593
	Furthest	43	69.25149	1043	70.31399
63	Nearest	49	70.54451	1049	71.60701
	Furthest	47	65.78542	1047	66.84792

APPENDIX B

INTERFERENCE DIMENSIONS FOR Davis-Besse, Unit1

FROM

Framatome Document 51-5013435-02 "CRDM Nozzle/Bore Dimensional Analysis"

	Revision	0			
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Table 3: Summary of Top and Bottom Dimensional Fits for CRDM Nozzles in the RV Closure Head for D-B

Penetration #	Dimensional Fit Top (in)	Dimensional Fit Bottom (in)	Penetration #	Dimensional Fit Top (in)	Dimensional Fit Bottom (in)
1	(0.0001)	(0.0012)	36	(0.0009)	(0.0002)
2	(0.0019)	(0.0020)	37	(0.0009)	(0.0010)
3	(0.0013)	(0.0015)	38	(0.0008)	(0.0007)
4	(0.0015)	(0.0016)	39	(0.0007)	(0.0008)
5	(0.0007)	(0.0009)	40	(0.0003)	(0.0010)
6	(0.0012)	(0.0008)	41	(0.0006)	(0.0009)
7	(0.0001)	(0.0009)	42	(0.0009)	(0.0005)
8	(0.0006)	(0.0008)	43	(0.0009)	(0.0009)
9	(0.0007)	(0.0008)	44	(0.0012)	(0.0002)
10	(0.0013)	(0.0002)	45	(0.0014)	(0.0011)
11	(0.0009)	(0.0006)	46	(0.0004)	(0.0012)
12	(0.0006)	(0.0003)	47	(0.0013)	(0.0007)
13	(0.0009)	0.0001	48	(0.0009)	(0.0013)
14	0.0004	0.0005	49	(0.0002)	0.0002
15	(0.0007)	(0.0006)	50	(0.0021)	(0.0010)
16	(0.0009)	(0.0009)	51	(0.0012)	(0.0018)
17	(0.0014)	(0.0013)	52	(0.0005)	(0.0009)
18	(0.0009)	(0.0007)	53	(0.0016)	(0.0013)
19	(0.0007)	(0.0004)	54	0.0000	0.0001
20	(0.0008)	(0.0008)	55	(0.0011)	(0.0010)
21	(0.0002)	0.0001	56	(0.0015)	(0.0011)
22	(0.0004)	(0.0008)	57	(0.0010)	(0.0006)
23	(0.0014)	(0.0006)	58	(0.0008)	(0.0005)
24	(0.0016)	0.0004	59	(0.0008)	(0.0001)
25	(0.0007)	(0.0012)	60	(0.0005)	(0.0011)
26	(0.0006)	(0.0008)	61	(0.0012)	(0.0003)
27	(0.0011)	(0.0005)	62	(0.0013)	(0.0004)
28	(0.0004)	(0.0008)	63	(0.0014)	(0.0015)
29	(0.0009)	(0.0010)	64	(0.0007)	(0.0005)
30	(0.0013)	(0.0011)	65	(0.0010)	(0.0004)
31	(0.0008)	(0.0010)	66	(0.0011)	(0.0012)
32	(0.0007)	0.0002	67	(0.0005)	(0.0006)
33	(0.0018)	(0.0003)	68	(0.0012)	(0.0013)
34	(0.0016)	(0.0010)	69	(0.0010)	(0.0005)
35	(0.0002)	0.0010			

	Revision	0			
	Preparer/Date	RLB 10-08-01			
	Checker/Date	STC 10-08-01			
	File No. W-ENTP-11Q-306			Page B2 of B2	

APPENDIX C

CASE STUDY FOR USE OF CONTAC52 ELEMENTS TO IMPOSE INTERFERENCE FIT LOADS

A comparison study was performed using the ANSYS software package [1] to verify that the CONTAC52 element would be an adequate means of simulating the interference load between the CRDM housing tube and the closure head. The verification consisted of two finite element models. The first would use the CONTAC52 gap element while the other would use a simple application of imposed displacement.

Model #1 – CONTAC52 Elements

The first model was a simple tube and plate model as shown in Figure C-1. The plate was 20"x20"x4" and was assumed to be infinitely rigid (Modulus of Elasticity, E = 30e12 psi).

A 4 inch diameter hole was centered through the plate into which was inserted a 3.998 inch diameter, 10 inch long tube that had a wall thickness of 0.4999 inches. The slight reduction in tube diameter was necessary to support the interference function CONTAC52 elements. The tube was inserted such that its base was flush with the bottom of the plate leaving 6 inches of the tube protruding from the top. The tube was modeled with a Modulus of Elasticity, E, of 30e6 psi.

CONTAC52 elements were applied at the plate to tube interface with an interference value of –0.01 inches. This should simulate the existence of a 0.01 inch radial interference between the tube and the plate throughout the circumference of the tube and the 4 inch thickness of the plate.

The plate edges and bottom along with the bottom of the tube are held with symmetry boundary conditions. See ANSYS input file TEST2.INP on the project CD-Rom.

The resulting stress intensity in the tube is shown in Figure C-2 and peaks at 233629 psi.

Model #2 – Imposed Displacements

The second model models only the tube from the previous model. Dimensions, mesh density and materials are all the same. The interference load for this analysis consisted of a series of imposed -0.01 inch radial displacements located at the same locations as the gap elements in the previous analysis.

The base of the tube was held with symmetric boundary conditions and a pair of opposing nodes was held in the circumferential direction to prevent rigid body motion. Figure C-3 showns the resulting model and the boundary conditions (including the imposed –0.01 inch radial displacements). See ANSYS input file TEST2a.INP on the project CD-Rom.

The resulting stress intensity for this load is shown in Figure C-4 and peaks at 233776 psi. The stress intensity for this load method is 0.0629% greater than the theoretical same load applied via the CONTAC52 elements. Clearly use of the CONTAC52 element for interference loading was acceptable.

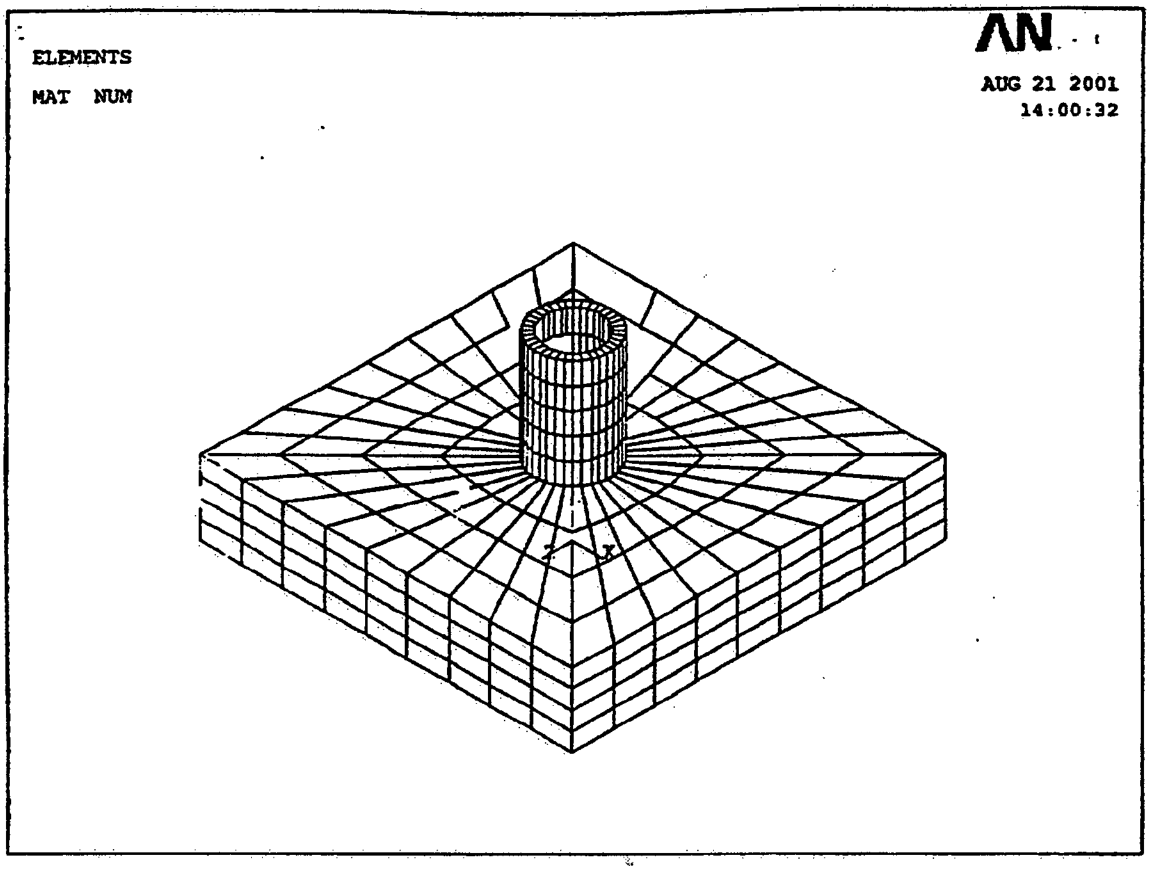


Figure C-1 – Finite Element Model Using Gaps

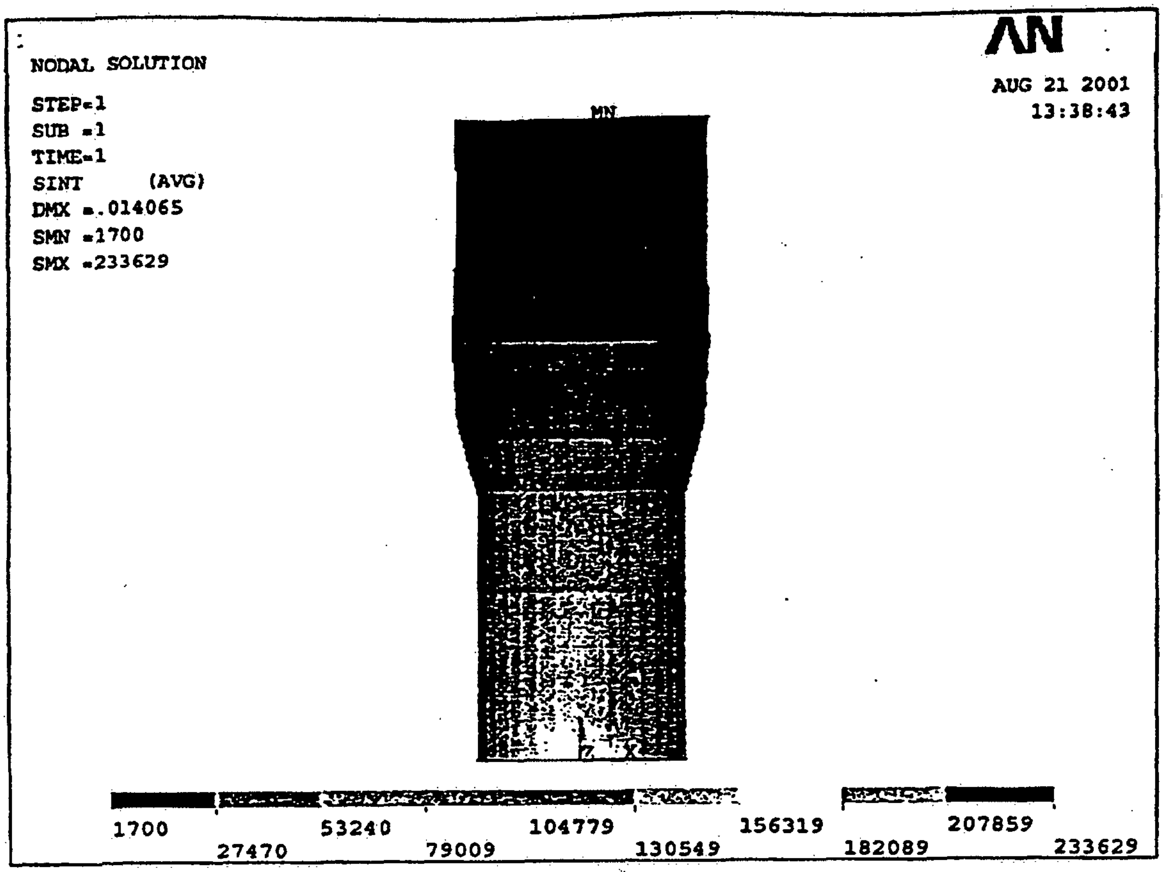


Figure C-2 – Stress Intensity in Tube for Gap Interference Analysis

	Revision	0			
	Preparer/Date	RLB 10-08-01			
	Checker/Date	STC 10-08-01			
	File No. W-ENTP-11Q-306			Page C4 of C6	

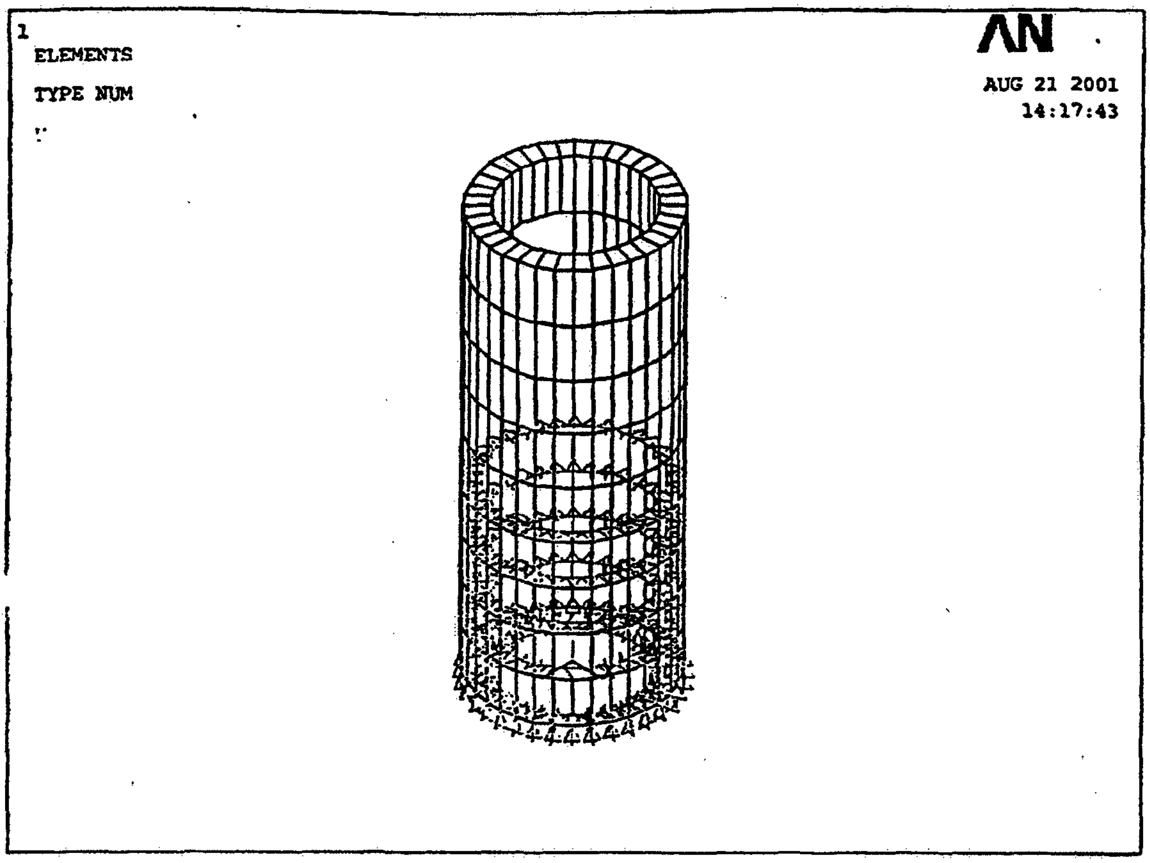


Figure C-3 – Finite Element Model with Boundary Conditions
Using Imposed Displacements for Loading

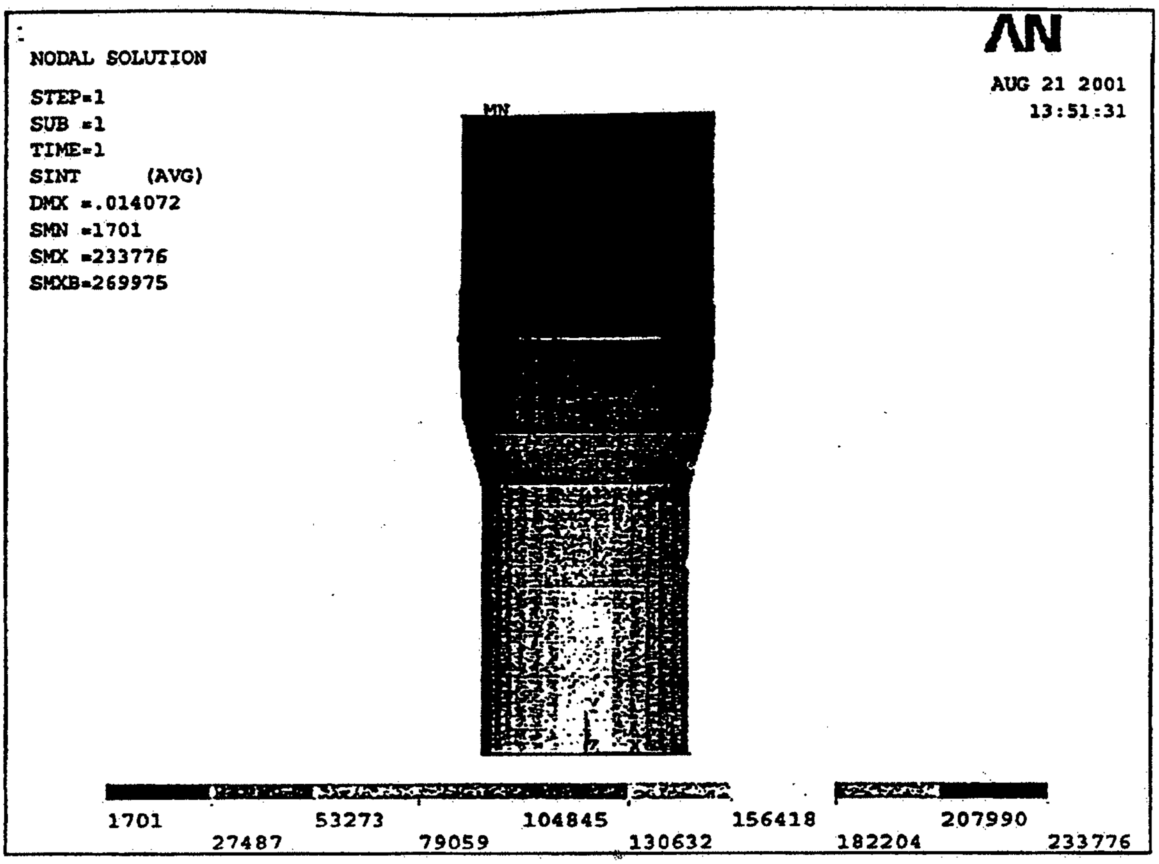


Figure C-4 – Stress Intensity in Tube for Imposed Displacement Analysis

	Revision	0			
	Preparer/Date	RLB 10-08-01			
	Checker/Date	STC 10-08-01			
	File No. W-ENTP-11Q-306			Page C6 of C6	

APPENDIX D

ADDITIONAL EVALUATIONS OF CRDM #3 GROUP INTERFERNCE CONDITIONS

	Revision	0			
	Preparer/Date	RLB 10-08-01			
	Checker/Date	STC 10-08-01			
	File No. W-ENTP-11Q-306			Page D1 of D3	

D.1.0 Objective

In the original evaluation, CRDM Tube 3 was included in the finite element model. Per Table 1 of Section 5.6, CRDM Tube 3 was grouped with CRDM Tubes 2, 4 and 5. Also in Table 1,CRDM Tube 3 was actually loaded with CRDM Tube 2's interference values, Tube 2 having been selected as the most conservative interference condition of the group for leak rate evaluations.

It was shown in Table 3, Section 6.1, that the Tube 3 had no gaps at the very bottom of the interference zone, resulting in no potential leakage. As a result, additional evaluations were performed in order investigate the other tubes in the group.

D.2.0 Finite Element Model Changes

The only changes that will be made to the original finite element model is to Tube 3's interference values. Table D-1 lists the affected tube group interference values. Interference values are from Reference 5 (see also Appendix B).

Table D-1
Interference Values For CRDM Tube 3 Group

Tube	Diametrical Interference Dimensions (in)	
	Top	Bottom
2	0.0019	0.0020
3	0.0013	0.0015
4	0.0015	0.0015
5	0.0007	0.0009

The original evaluation used Tube 2 interference values. Two additional evaluations will be performed. The first will use Tube 4 interference values and is conservatively considered to include Tube 3 since the area of concern is the bottom portion of the interference zone. The second evaluation will use Tube 5 interference values. The modified ANSYS input files for these two evaluations are named DB4CRDM.INP and DB5CRDM.INP (both are included on the project CD-ROM).

D.3.0 Gap Opening Evaluation Results

For the evaluation using Tube 4 interference values, it was once again determined that there were no gaps at the very bottom of the interference zone (similar to the original evaluation results). The gap data for the evaluation is included in the Excel spreadsheet APPEN_D.XLS (included on the project CD-Rom).

For the evaluation using Tube 5 interference values, it was determined that several gaps were now available through the interference zone.

Table D-2 lists the smallest gap opening along the vertical path of modeled Tube 3 for both interference cases described in the previous page. In the case of the Tube 4 interference case where there was no gap, the minimum interference value is listed in the form of a negative value. A complete list of tube results for the Tube 4 and 5 cases can be found in the Excel spreadsheet APPEN_D.XLS (included on the project CD-Rom).

Table D-2
Minimum Gap Results for Tube 3 Leakage Evaluations

Interference Value Used Tube Number	Minimum Gap (inches)
2 (Original Case)	-0.00002483 (Interference)
4	-0.00000942 (Interference)
5	0.000074799

D.4.0 Conclusions

Based on the results of these additional evaluations, it has been determined that Tubes 1, 2, 3 and 4 provide no gap through which leakage may occur during normal operating conditions. All other tubes have gaps through which leakage may occur.

Revision	0			
Preparer/Date	RLB 10-08-01			
Checker/Date	STC 10-08-01			

NRC LETTERS - REVIEW AND APPROVAL REPORT
ED 7159-7

RASC-153

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 12
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

(1) RECORD RA

(3) SUMMARY (Log No., Title Subject)
Supplemental Information in Response

(4) COMMITMENT LIST ADDED TO LETTER ☒

(5) PERIODIC ☐ YES

(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC
10/17/01 (Target) ☒ N/A

(7) SPECIAL ☒ EXPRE

(8) PREPARED BY
RMCook, et al.

(9) NOTARY ☒ YES ☐

(11) ADDITIONAL REFERENCES

(12) ... CLOSED

DOCKETED
USNRC
September 9, 2009 (11:00am)
OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

(13) COMMENTS

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ COGNIZANT INDIVIDUAL	RMCook	[signature]	10/17/01	10/17/01
☒ RESPONSIBLE SUPERVISOR	DLMiller	Dcm	10/12/01	10/12/01
☒ RESPONSIBLE MANAGER – DBE	DGeisen	[signature]	10/17/01 DEW	10/17/01
☐ PLANT MANAGER				
☒ Design Basis Engrg-Mech	PGoyal	[signature]	10/17/01 DEW	10/17/01
☒ Technical Services – Director	SPMoffitt	[signature]	10/17/01	10/17/01
☒ Plant Engrg – Systems Engr.	ASiemaszko	AS.	10/17/01 DEW	10/17/01
☒ Proj. Mgr	MMcLaughlin	[signature]	10/17/01 DEW	10/17/01
☐				
☒ SUPERVISOR	DRWuokko	[signature]	10/17/01	10/17/01
☒ MANAGER	DHLockwood	[signature]	10-17-01	10-17-01
☒ VICE PRESIDENT	GG Campbell	[signature]	10/17/01	10/17/01

DATE ADDED TO LETTER ☐ 10-17-01
(15) DATE ADDED BY Laura A. Jennison
(17) ADDITIONAL DISTRIBUTION

DATE SENT TO NRC 10-18-01
(16) DISTRIBUTED BY Laura A. Jennison

DATE OF BLIND DISTRIBUTION 10-22-01
(18) DISTRIBUTED BY Laura A. Jennison

S11-00598

TEMPLATE = SECY 028 DS02

NRC027-1698

NRC LETTERS - REVIEW AND APPROVAL REPORT
ED 7159-7

(1) RECORDS MANAGEMENT NO. RAS-01-00446	(2) SERIAL NO. 2735

(3) SUMMARY (Log No., Title Subject): Supplemental Information in Response to NRC Bulletin 2001-01

(4) COMMITMENT LIST ADDED TO LETTER ☒

(5) PERIODIC / NON-PERIODIC REPORT ☐ YES ☒ NO REPORT NO.

(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC: 10/17/01 (Target) ☒ N/A

7) SPECIAL HANDLING ☒ EXPRESS DELIVERY ☐ TELECOPY — DATE SENT 10/17/01

(8) PREPARED BY: RMCook, et al.

(9) NOTARY ☒ YES ☐ NO

(10) LICENSE FEE REQUIRED ☐ YES ☒ NO

(11) ADDITIONAL REFERENCES

(12) COMMITMENT NO.(S) CLOSED

(13) COMMENTS

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ COGNIZANT INDIVIDUAL	RMCook	[signature]	10/17/01	10/17/01
☒ RESPONSIBLE SUPERVISOR	DLMiller	DCM	10/12/01	10/12/01
☒ RESPONSIBLE MANAGER - DBE	DGeisen	[signature]	10/17/01 DEW	10/17/01
☐ PLANT MANAGER				
☒ Design Basis Engrg-Mech	PGoyal	[signature]	10/17/01 DEW	10/17/01
☒ Technical Services - Director	SPMoffitt	SPMoffitt	10/17/01	10/17/01
☒ Plant Engrg - Systems Engr.	ASiemaszko	AS.	10/17/01 DEW	10/17/01
☒ Proj. Mgr	MMcLaughlin	[signature]	10/17/01 DEW	10/17/01
☐				
☒ SUPERVISOR	DRWuokko	[signature]	10/17/01	10/17/01
☒ MANAGER	DHLockwood	[signature]	10-17-01	10-17-01
☒ VICE PRESIDENT	GGCampbell	GGC	10/17/01	10/17/01

DATE ADDED TO LETTER ☐ 10-17-01

(15) DATE ADDED BY: Laura A. Jennison

DATE SENT TO NRC: 10-18-01

(16) DISTRIBUTED BY: Laura A. Jennison

DATE OF BLIND DISTRIBUTION: 10-22-01

(18) DISTRIBUTED BY: Laura A. Jennison

(17) ADDITIONAL DISTRIBUTION

S11-00598

NRC027-1698

The NRC Letters - Review and Approval Report (ED 7159-7) should be completed by the Regulatory Affairs Section.

BLOCK 1	RECORDS MANAGEMENT NO. - Regulatory Affairs enters Records Management number prior [illegible] distribution of correspondence to NRC.
BLOCK 2	SERIAL NO. - Initiator enters serial number obtained from the Regulatory Affairs Clerk.
BLOCK 3	SUMMARY (Log No., Title Subject) - Initiator enters a summary of the correspondence. This summary should identify if the correspondence is in response to any previous correspondence and why the letter i being written.
BLOCK 4	COMMITMENT LIST ADDED TO LETTER - Preparers checks the block to indiate a commitment list has been included with the letter.
BLOCK 5	PERIODIC/NON-PERIODIC REPORT - Identify whether this correspondence is a Periodic or Non-Periodic Report as identified in Nuclear Group Procedure NG-NS-00807.
BLOCK 6	DATE RESPONSE DUE TO BE SUBMITTED TO NRC - Initiator enters the date the correspondence is due to the NRC. If the correspondence does not have a required due date, the block shall be marked not applicable (NA).
BLOCK 7	SPECIAL HANDLING - Initiator checks if the correspondence requires special distribution to the NRC. I yes, the Regulatory Affairs clerk enters date the correspondence is sent.
BLOCK 8	PREPARED BY - Initiator enters the names of individuals responsible for providing technical information for the correspondence along with his/her name.
BLOCK 9	NOTARY - Initiator checks if the correspondence is required to be notarized.
BLOCK 10	LICENSE FEE REQUIRED - Initiator checks if a license fee is required, per the requirements of [illegible]R Part 170. If yes, the initiator shall complete a Voucher Check Authorization (Form 294) and obt[illegible] appropriate fees to accompany the correspondence.
BLOCK 11	ADDITIONAL REFERENCES - Initiator enters any additional NRC correspondence or documents that pertain to the subject correspondence.
BLOCK 12	COMMITMENT NO(S). CLOSED - Initiator enters the Commitment Management System number(s) of any commitments that are closed by the subject correspondence.
BLOCK 13	COMMENTS - Initiator or any reviewer enters appropriate comments regarding the subject correspondence.
BLOCK 14	REVIEW AND APPROVAL - Initiator checks and /or enters the desired reviewer(s) . The technical accuracy of a response to the NRC is the responsibility of the Director and Management individual assigned the action.
BLOCK 15	DATE ADDED BY - Distributor checks the Date Added to Letter Block and signs the Date Added By block to indicate the original letter has been dated prior to distribution to the NRC.
BLOCK 16	DISTRIBUTED BY - Distribution to the NRC shall be made by the Regulatory Affairs Section. Distributo signs the Distributed By block and completes the Date Sent to NRC block.
BLOCK 17	ADDITIONAL DISTRIBUTION - Initiator enters individuals requiring distribution that are not on the standard distribution list.
BLOCK 18	DISTRIBUTED BY - Distributor signs the Distributed By block and completes the Date of Blind Distribution block.

S11-00599

RAS C.154

DOCKETED
USNRC

September 9, 2009 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

FirstEnergy

Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449-9760

G. Campbell
e President - Nuclear

419-321-8588
Fax: 419-321-8337

Attachment Contains Restricted
Material Per 10 CFR 2.790

Docket Number 50-346

License Number NPF-3

Serial Number 2744

October 30, 2001

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 13
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Transmittal of Results of Reactor Pressure Vessel Head Control Rod Drive Mechanism Nozzle Penetration Visual Examinations for the Davis-Besse Nuclear Power Station

Ladies and Gentlemen:

During a public meeting between the Davis-Besse Nuclear Power Station (DBNPS) staff and the Nuclear Regulatory Commission (NRC) staff on October 24, 2001, concerning the FirstEnergy Nuclear Operating Company (FENOC) response (letter Serial Number 2731, dated September 4, 2001) to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," the DBNPS staff committed to provide pictorial documentation of the visual examinations of the reactor pressure vessel head performed during the DBNPS 10th, 11th and 12th refueling outages. This documentation is provided in Attachment 1 as the DBNPS report, "Results of Visual Examination of Reactor Head CRDM Nozzle Penetration Performed During 1996, 1998, and 2000."

This report is considered to be restricted by the FENOC and is requested to be withheld from public disclosure pursuant to 10 CFR 2.790. An affidavit complying with the requirements of 10 CFR 2.790 is provided in Attachment 2 citing the basis for this report to be withheld from public disclosure.

The inspections performed during the 10th, 11th, and 12th Refueling Outage (10RFO, conducted April 8 to June 2, 1996; 11RFO, conducted April 10, to May 23, 1998; and,

AP01

TEMPLATE = SECY-028

DS03

12RFO, conducted April 1 to May 18, 2000) consisted of a whole head visual inspection of the RPV head in accordance with the DBNPS Boric Acid Corrosion Control Program pursuant to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The visual inspections were conducted by remote camera and included below insulation inspections of the RPV bare head such that the Control Rod Drive Mechanism (CRDM) nozzle penetrations were viewed. During 10RFO, 65 of 69 nozzles were viewed, during 11RFO, 50 of 69 nozzles were viewed, and during 12RFO, 45 of 69 nozzles were viewed. It should be noted that 19 of the obscured nozzles in 12RFO were also those obscured in 11RFO. Following 11RFO, the RPV head was mechanically cleaned in localized areas as limited by the service structure design. Following 12RFO, the RPV head was cleaned with demineralized water to the extent possible to provide a clean head for evaluating future inspection results.

The affected areas of accumulated boric acid crystal deposits were video taped, and have subsequently been reviewed with specific focus on boric acid crystal deposits with reference to the CRDM nozzle penetration leakage as previously observed at the Oconee Nuclear Station, Unit 3 (ONS-3) and at Arkansas Nuclear One, Unit 1 (ANO-1). During the 12RFO inspection, 24 of the 69 nozzles were obscured by boric acid crystal deposits that were clearly attributable to leaking motor tube flanges from the center CRDMs. A further subsequent review of the video tapes has been conducted and the results of this review did not identify any boric acid crystal deposits that would have been attributed to leakage from the CRDM nozzle penetrations, but were indicative of CRDM flange leakage.

The aforementioned video taped images of areas of accumulated boric acid crystal deposits have been converted to photographic images and are contained in the attached report.

If you have any questions or require further information, please contact Mr. David H. Lockwood, Manager-Regulatory Affairs at (419) 321-8450.

Very truly yours,

Enclosure
Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, DB-1 NRC/NRR Project Manager
D. S. Simpkins, DB-1 Acting Senior Resident Inspector
Utility Radiological Safety Board

SUPPLEMENTAL INFORMATION

IN RESPONSE TO

NRC BULLETIN 2001-01

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f) and contains supplemental information concerning the response (Serial 2731, dated September 4, 2001) to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for the Davis-Besse Nuclear Power Station, Unit Number 1.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: Guy G. Campbell
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me 30th day of October, 2001.

Nora L. Flood
Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

Docket Number 50-346
License Number NPF-3
Serial Number 2744
Attachment 1
Page 1 of 1

Davis-Besse Nuclear Power Station

Results of Visual Examination of Reactor Head CRDM Nozzle Penetrations
Performed in 1996, 1998, and 2000

(46 Pages Follow)

Davis-Besse NPS

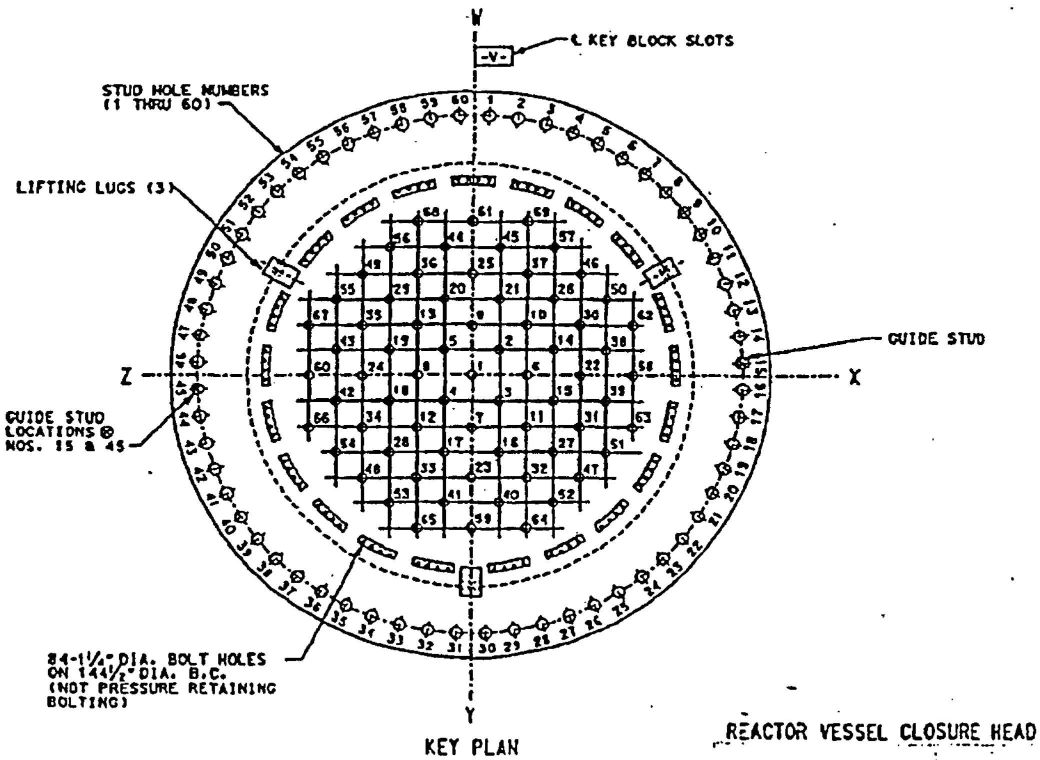
Results of Visual Examination

of

Reactor Head CRDM nozzle penetrations

Performed in 1996, 1998, and 2000

- Nozzle Arrangement – 69 CRDM Nozzles



Nozzle No.	Core Locat.	Quadrant	1996 Inspection results	1998 Inspection results	2000 Inspection results
			See Note 1.0		
1	H8	1		*Flange Leak Evident*	*Flange Leak Evident*
2	G7	4		*Flange Leak Evident*	*Flange Leak Evident*
3	**G9**	**1**		***Flange Leak Evident***	***Flange Leak Evident***
4	K9	2		*Flange Leak Evident*	*Flange Leak Evident*
5	K7	3		Flange Leak Evident	Flange Leak Evident
6	**F8**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
7	H10	2		Flange Leak Evident	Flange Leak Evident
8	L8	3		No Leak Observed	No Leak Observed
9	H6	4		No Leak Observed	No Leak Observed
10	F6	4		No Leak Observed	No Leak Observed
11	**F10**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
12	L10	2		No Leak Observed	No Leak Observed
13	L6	3		No Leak Recorded	No Leak Observed
14	E7	4		Flange Leak Evident	Flange Leak Evident
15	E9	1		Flange Leak Evident	Flange Leak Evident
16	G11	1		Flange Leak Evident	Flange Leak Evident
17	K11	2		No Leak Observed	No Leak Observed
18	M9	2		No Leak Recorded	No Leak Observed
19	M7	3		No Leak Observed	No Leak Recorded
20	K5	3		No Leak Observed	No Leak Observed
21	G5	4		No Leak Observed	No Leak Observed
22	D8	1		Flange Leak Evident	Flange Leak Evident
23	H12	2		No Leak Observed	No Leak Observed
24	N8	3		No Leak Recorded	No Leak Recorded
25	H4	4		No Leak Recorded	No Leak Observed
26	E5	4		No Leak Recorded	No Leak Observed
27	E11	1		Flange Leak Evident	Flange Leak Evident
28	M11	2		No Leak Recorded	No Leak Observed
29	M5	3		No Leak Recorded	No Leak Observed
30	D6	4		No Leak Observed	No Leak Observed
31	**D10**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
32	F12	1		Flange Leak Evident	Flange Leak Evident
33	L12	2		No Leak Recorded	No Leak Observed
34	N10	2		No Leak Recorded	No Leak Observed
35	N6	3		No Leak Recorded	No Leak Recorded
36	L4	3		No Leak Recorded	No Leak Observed
37	F4	4		No Leak Recorded	No Leak Observed
38	C7	4		No Leak Recorded	Flange Leak Evident
39	C9	1		Flange Leak Evident	Flange Leak Evident
40	G13	1		Flange Leak Evident	Flange Leak Evident
41	K13	2		No Leak Recorded	No Leak Observed
42	O9	2		No Leak Recorded	No Leak Recorded
43	O7	3		No Leak Recorded	No Leak Recorded
44	K3	3		No Leak Recorded	No Leak Observed
45	G3	4		No Leak Recorded	No Leak Observed
46	D4	4		No Leak Recorded	No Leak Observed
47	D12	1		Flange Leak Evident	Flange Leak Evident

Nozzle No.	Core Locat.	Quadrant	1996 Inspection results	1998 Inspection results	2000 Inspection results
48	N12	2		No Leak Recorded	No Leak Observed
49	N4	3		No Leak Recorded	No Leak Observed
50	C5	4		No Leak Recorded	No Leak Observed
51	**C11**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
52	E13	1		No Leak Recorded	Flange Leak Evident
53	M13	2		No Leak Recorded	No Leak Observed
54	O11	2		No Leak Recorded	No Leak Observed
55	O5	3		No Leak Recorded	No Leak Recorded
56	M3	3		No Leak Recorded	No Leak Observed
57	E3	4		No Leak Recorded	No Leak Observed
58	B8	1		No Leak Recorded	Flange Leak Evident
59	H14	2		No Leak Recorded	No Leak Observed
60	P8	3		No Leak Recorded	No Leak Recorded
61	H2	4		No Leak Recorded	No Leak Observed
62	B6	4		No Leak Recorded	No Leak Observed
63	B10	1		No Leak Recorded	Flange Leak Evident
64	F14	1		No Leak Recorded	Flange Leak Evident
65	L14	2		No Leak Recorded	No Leak Observed
66	P10	2		No Leak Recorded	No Leak Recorded
67	P6	3		No Leak Recorded	No Leak Recorded
68	L2	3		No Leak Recorded	No Leak Observed
69	F2	4		No Leak Recorded	No Leak Observed

Filed as h/RCS leakage issues/nozzle review Table

Notes:

1 In 1996 during 10 RFO, 100% of nozzles were inspected by visual examination. Since the video was void of head orientation narration, each specific nozzle view could not be correlated by nozzle number. Nozzles 1,2,3, and 4 which do not have sufficient interference gap were excluded. The remaining 65 nozzles did not show any evidence of leakage.

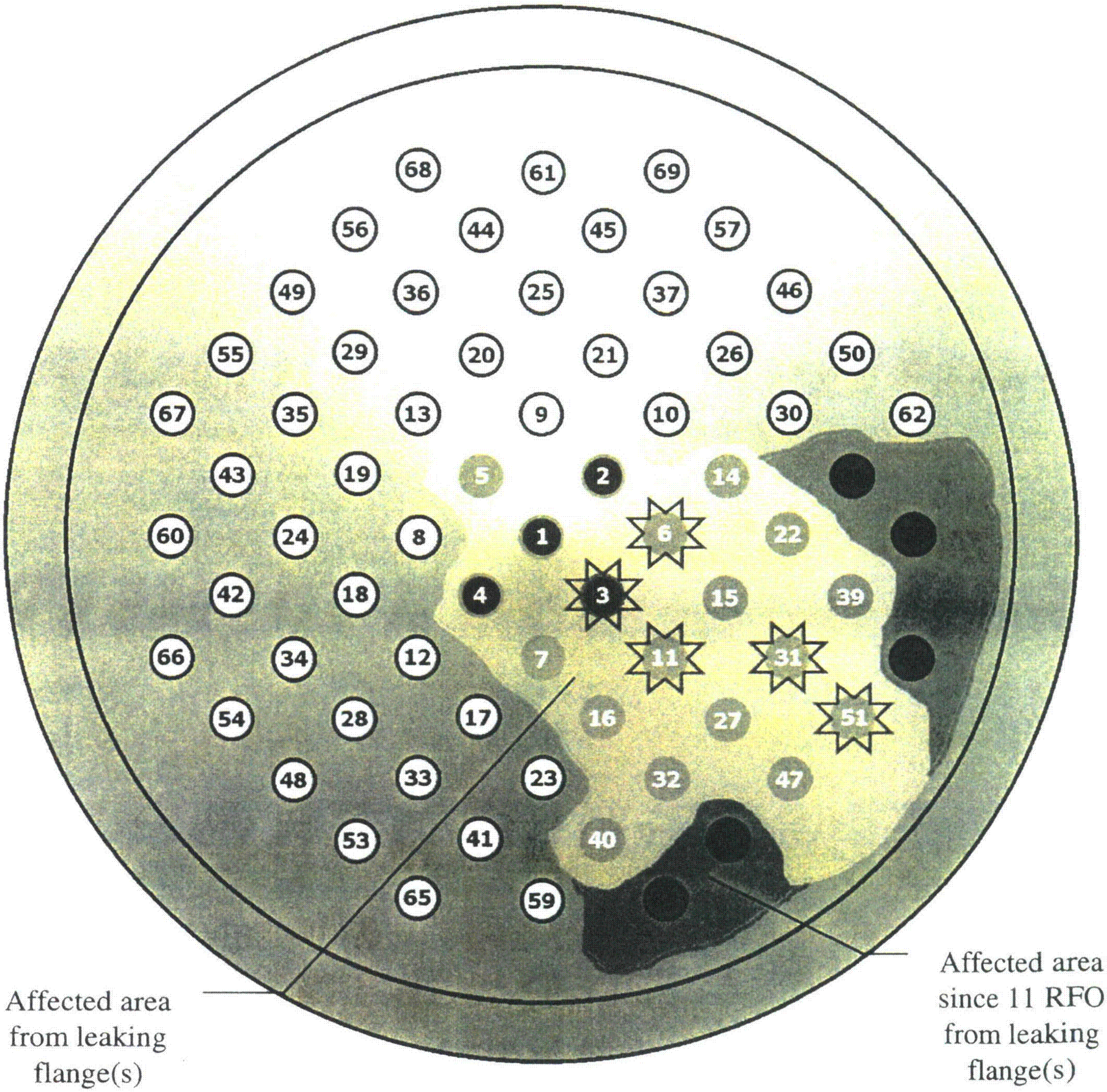
Bold letters indicate leaking CRDM bolting flanges discovered and repaired during 12 RFO (April 2000).

No Leak Observed = Visual Inspection Satisfactory, No Video Record Required.

No Leak Recorded = Nozzle inspection recorded on videotape

Italicized text indicates nozzles that are not expected to show leakage due to insufficient gap.

RPV Head 11 & 12 RFO Inspection Results



- (61) - No leakage identified
- (4) - Evaluated not to have sufficient gap to exhibit leakage
- (3) - Insufficient gap with leaking flange
- (5) - Nozzle obscured by boron
- (6) - Nozzle obscured by boron with leaking flange
- ● - Newly affected, since 11 RFO, by leaking flange(s)

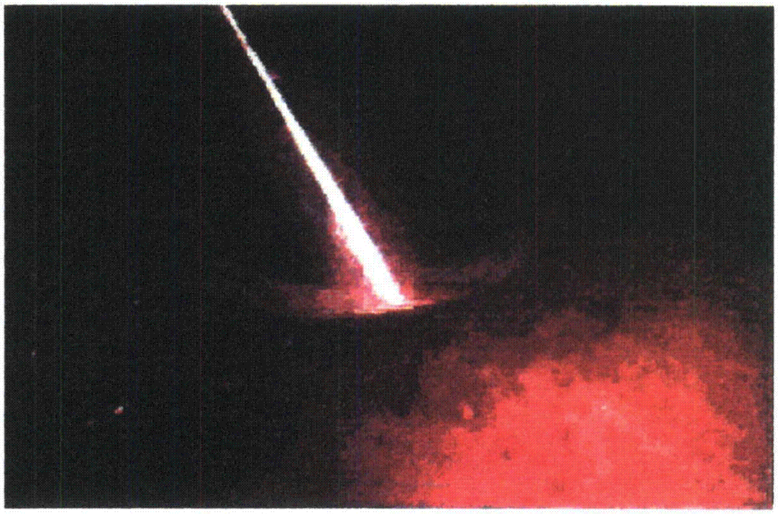
FENOC RESTRICTED INFORMATION

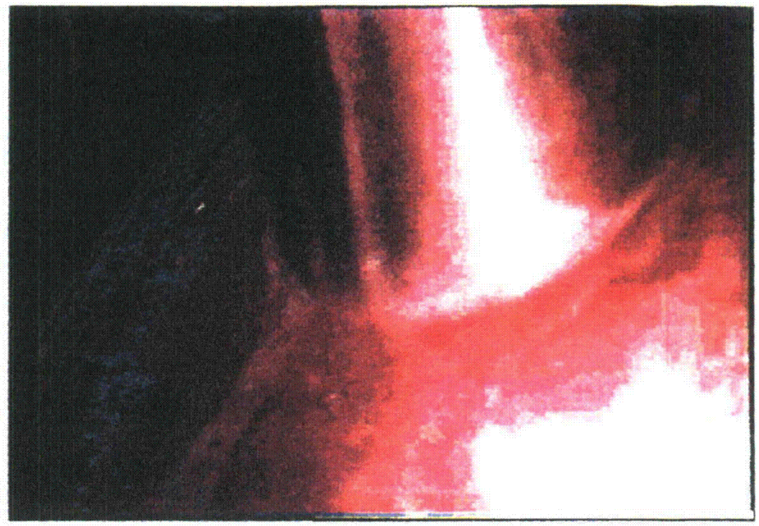
C61

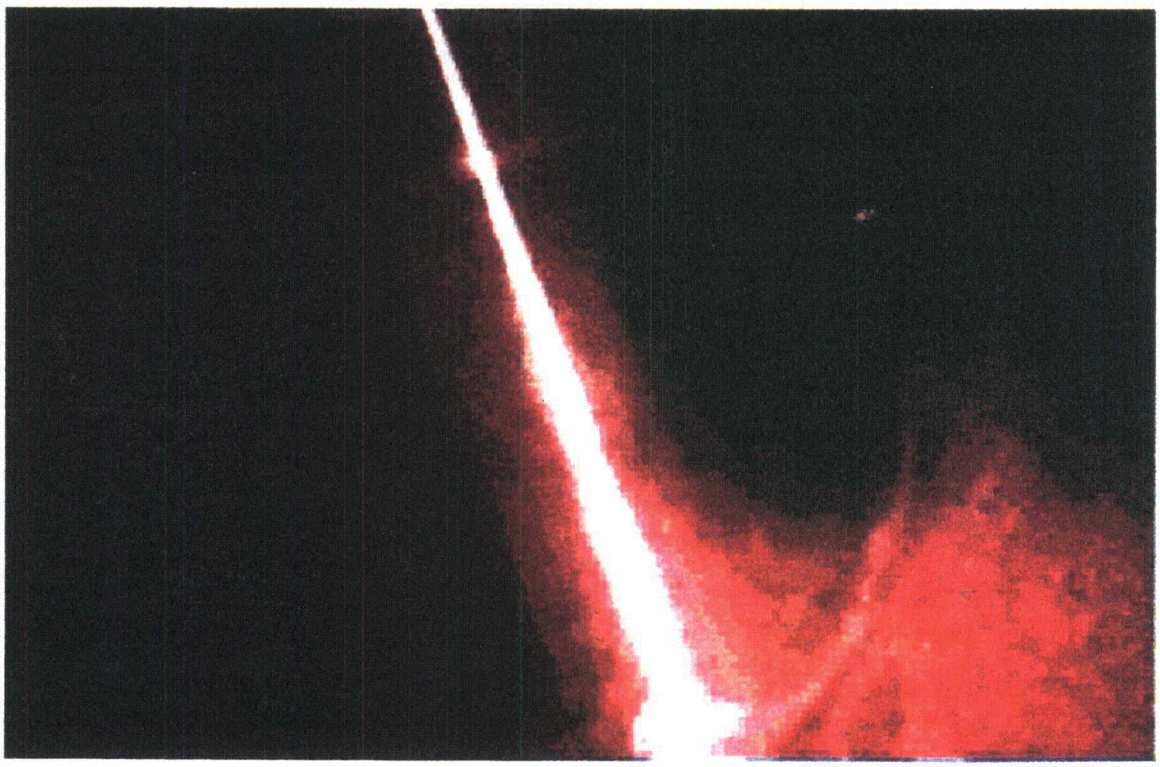
Spring 1996 Inspection

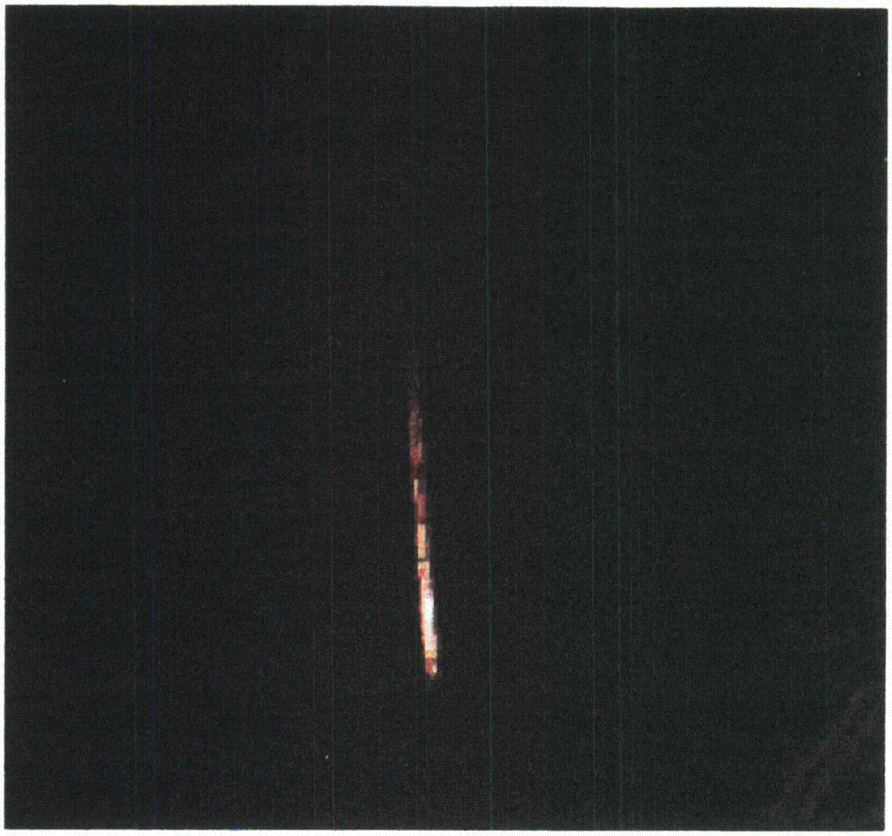
1996 Inspections

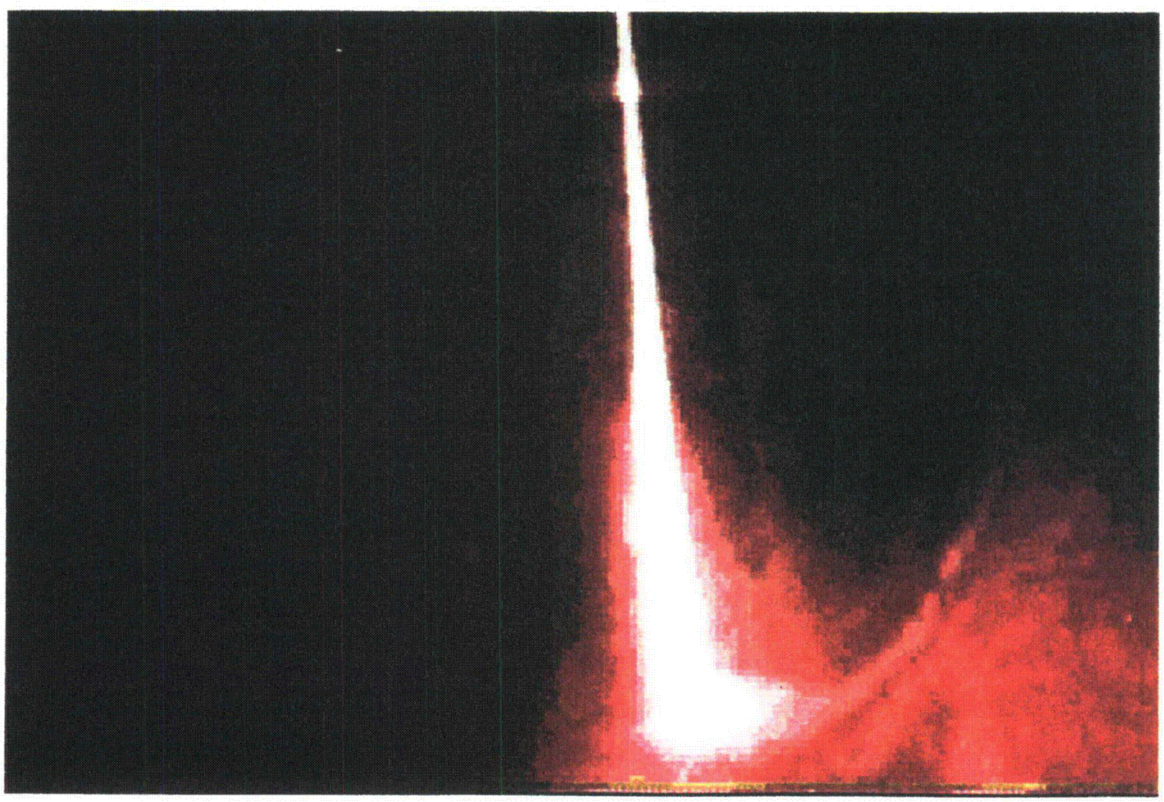
The following pictures are representative of the head in the Spring 1996 Outage. The head was relatively clean and afforded a generally good inspection.

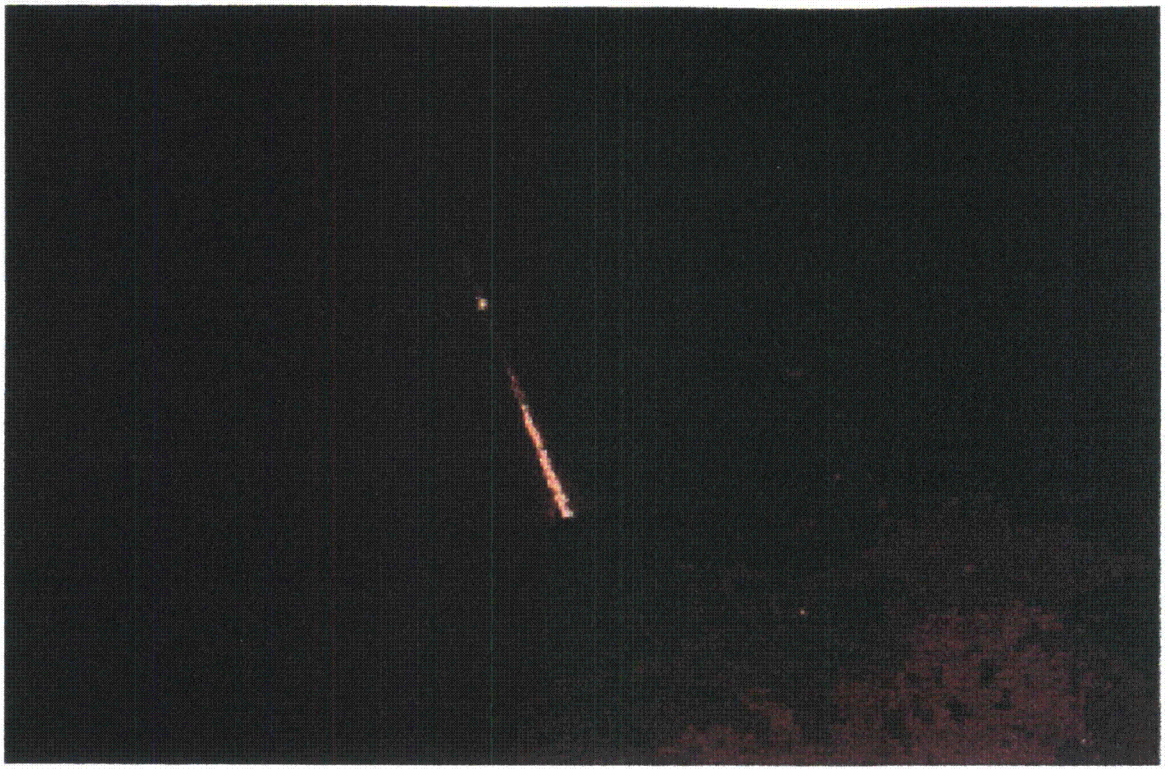






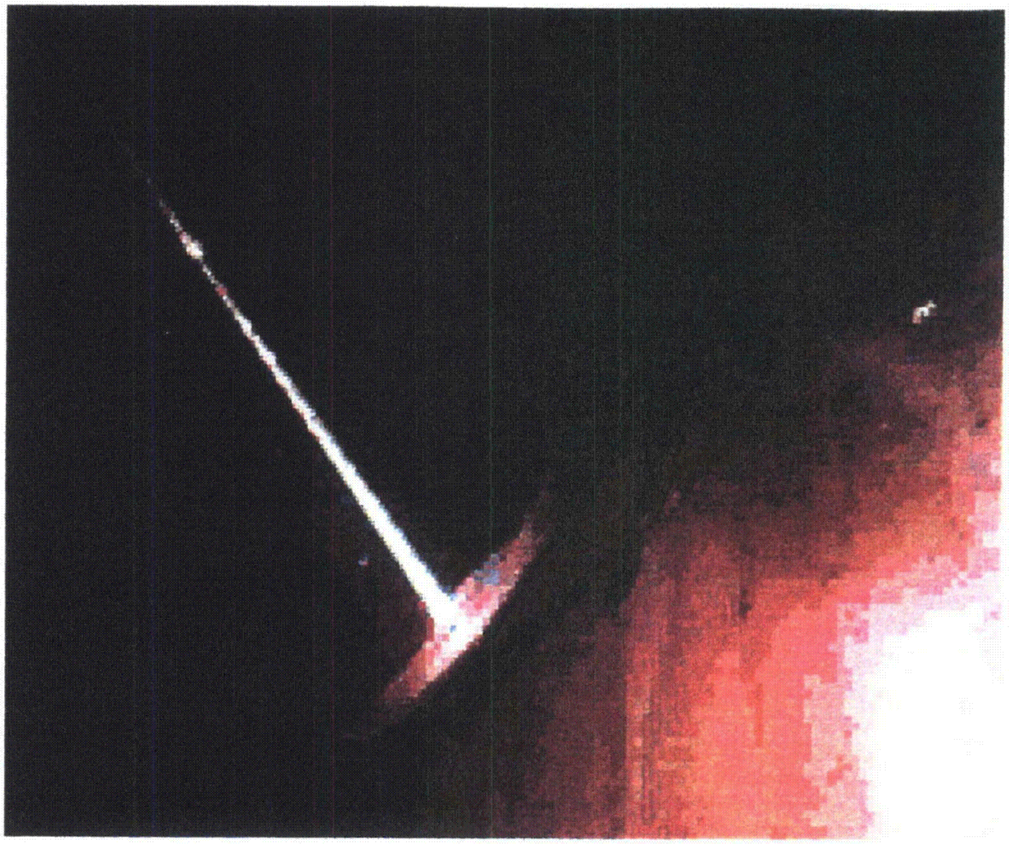




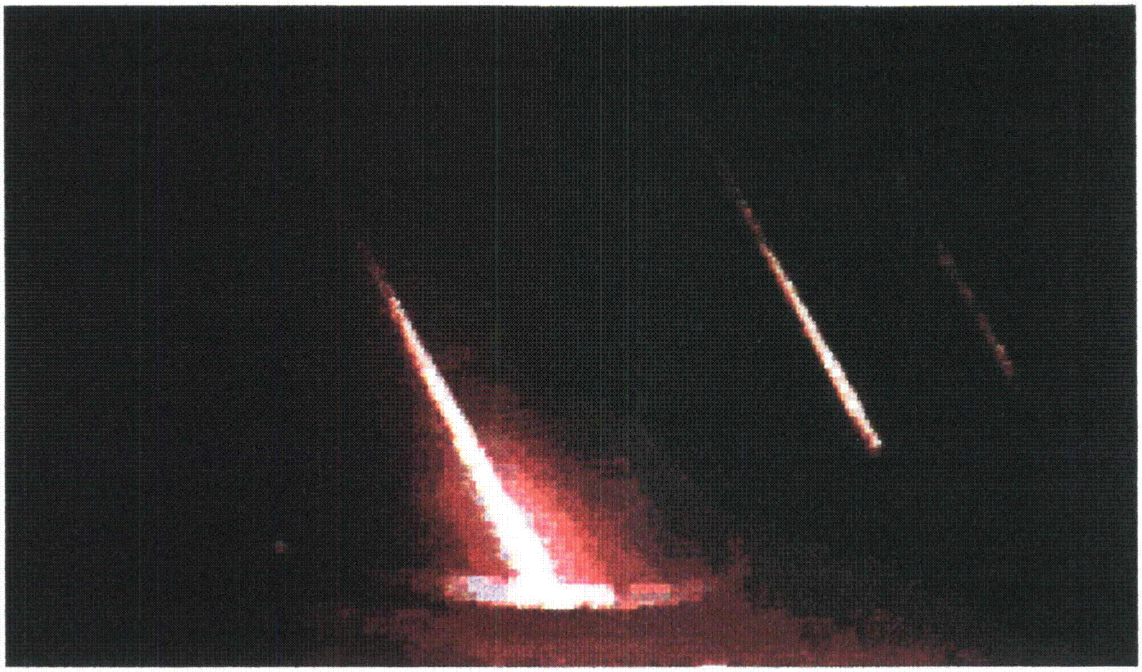


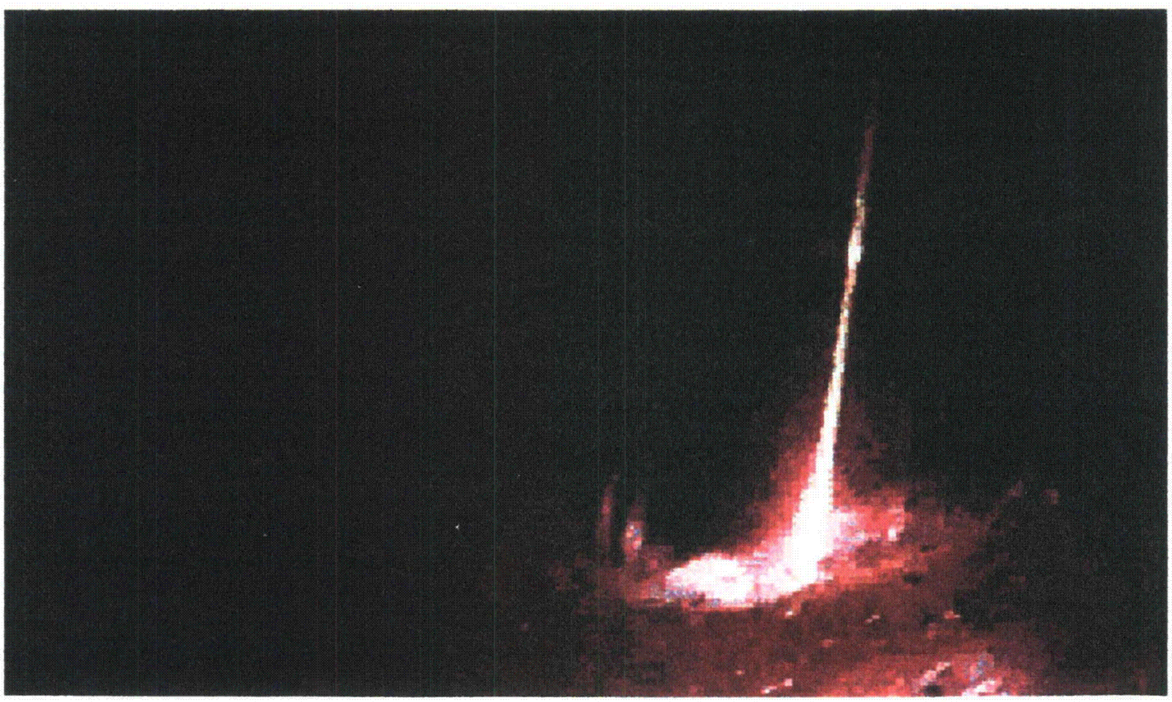
Some boron piles were observed at the top of the head in the vicinity of previous leaking flanges. Because of its location on the head, it could not be removed by mechanical cleaning but was verified to not be active or wet and therefore did not pose a threat to the head from a corrosion standpoint. Additionally, since these drives are not credited with leaking, that further ratifies that the boron is from previous flange leakage. The boron was heaviest beneath the mirror insulation seams.

1996 Inspections

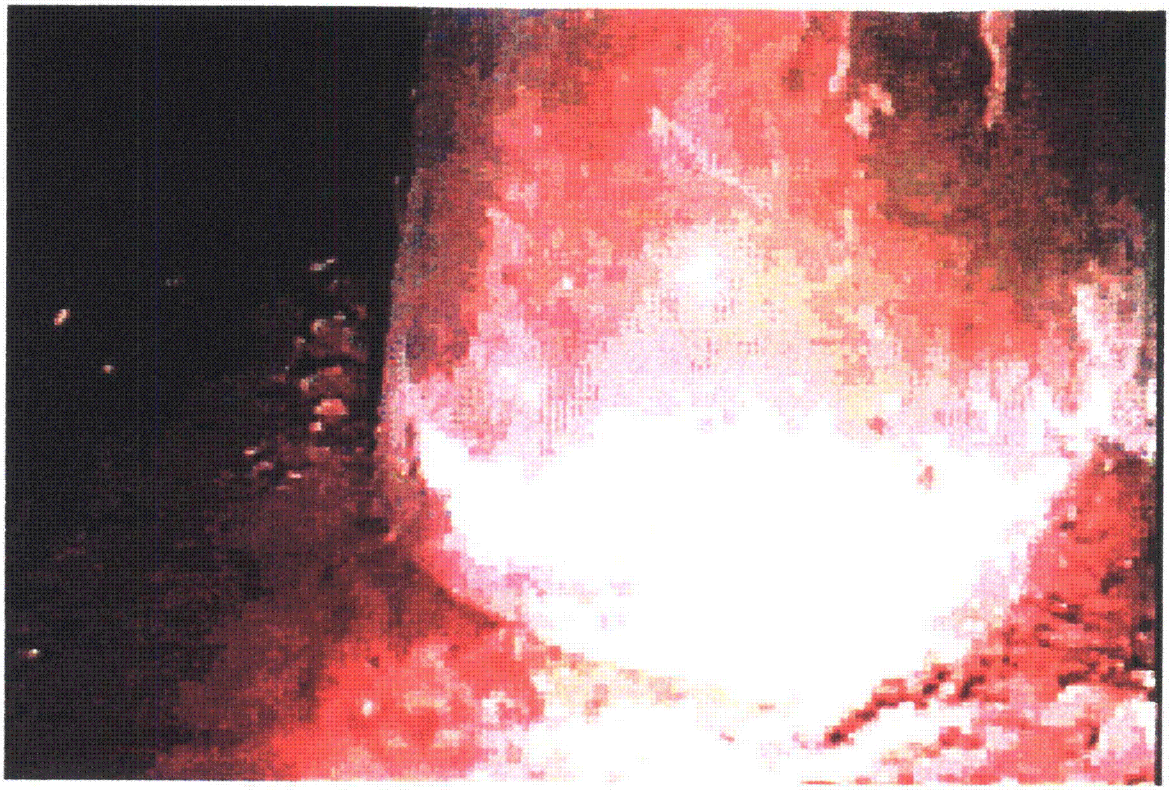


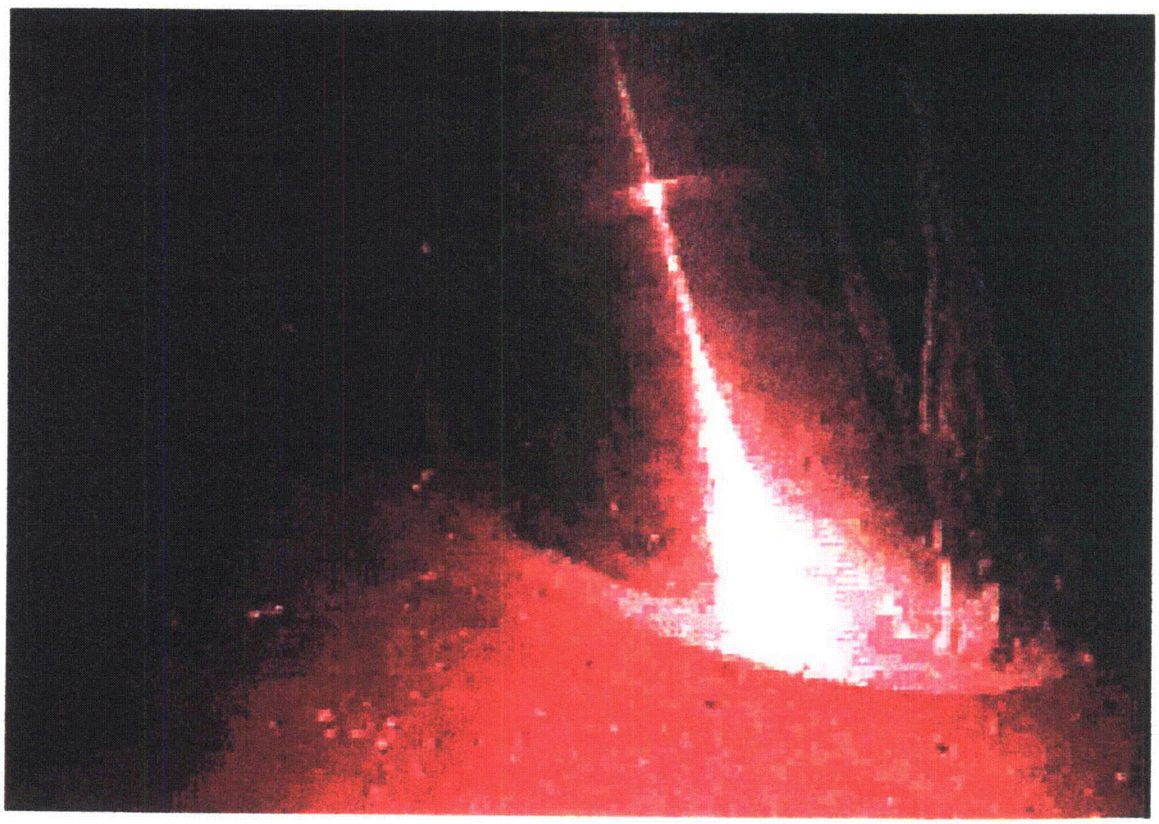
Hole 2



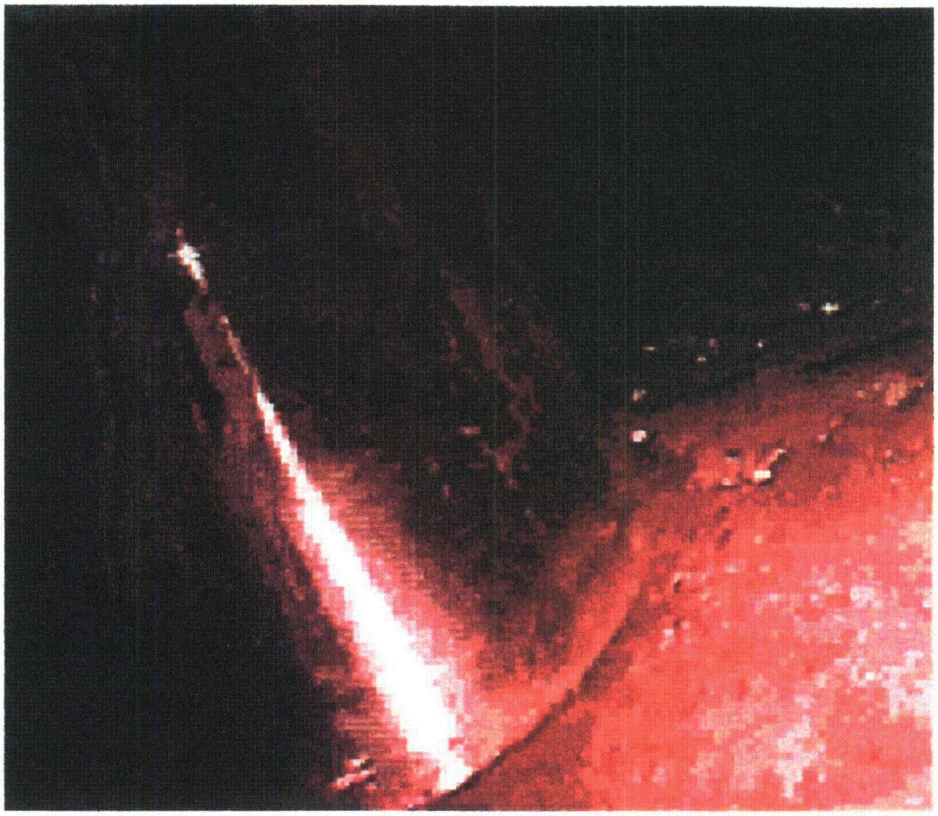


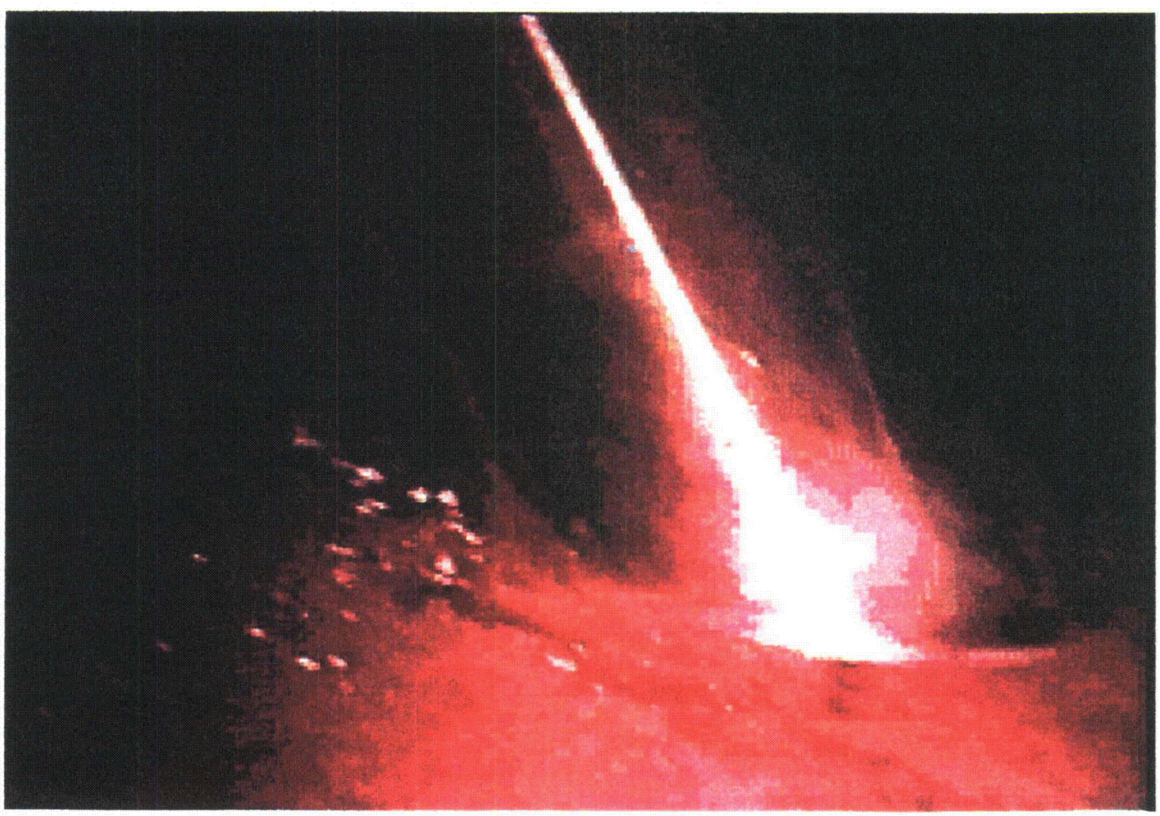
C06





FENOC RESTRICTED INFORMATION

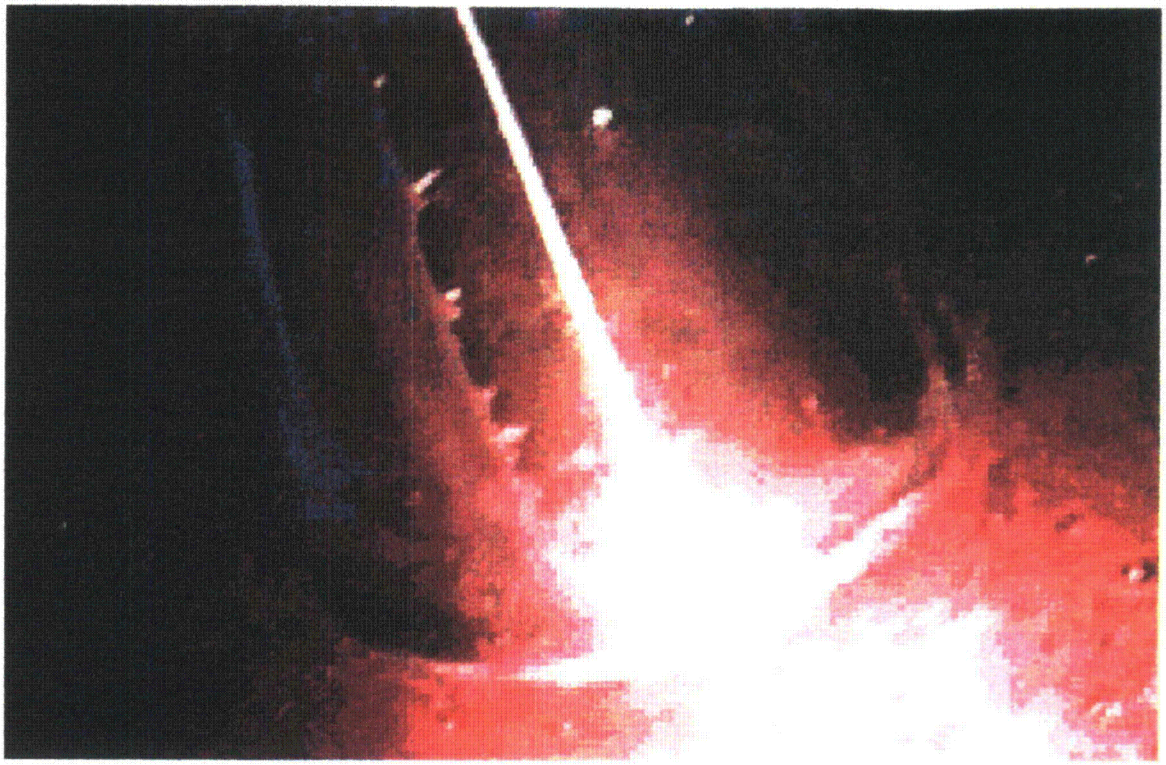


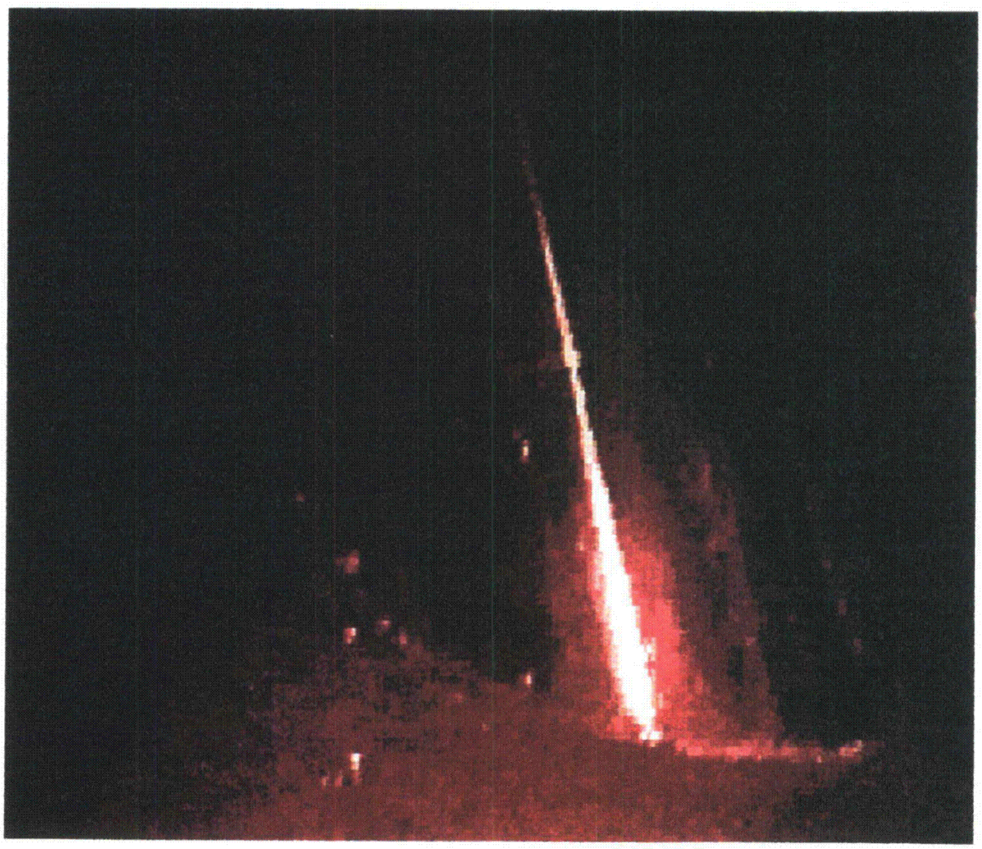


FENOC RESTRICTED INFORMATION

C08

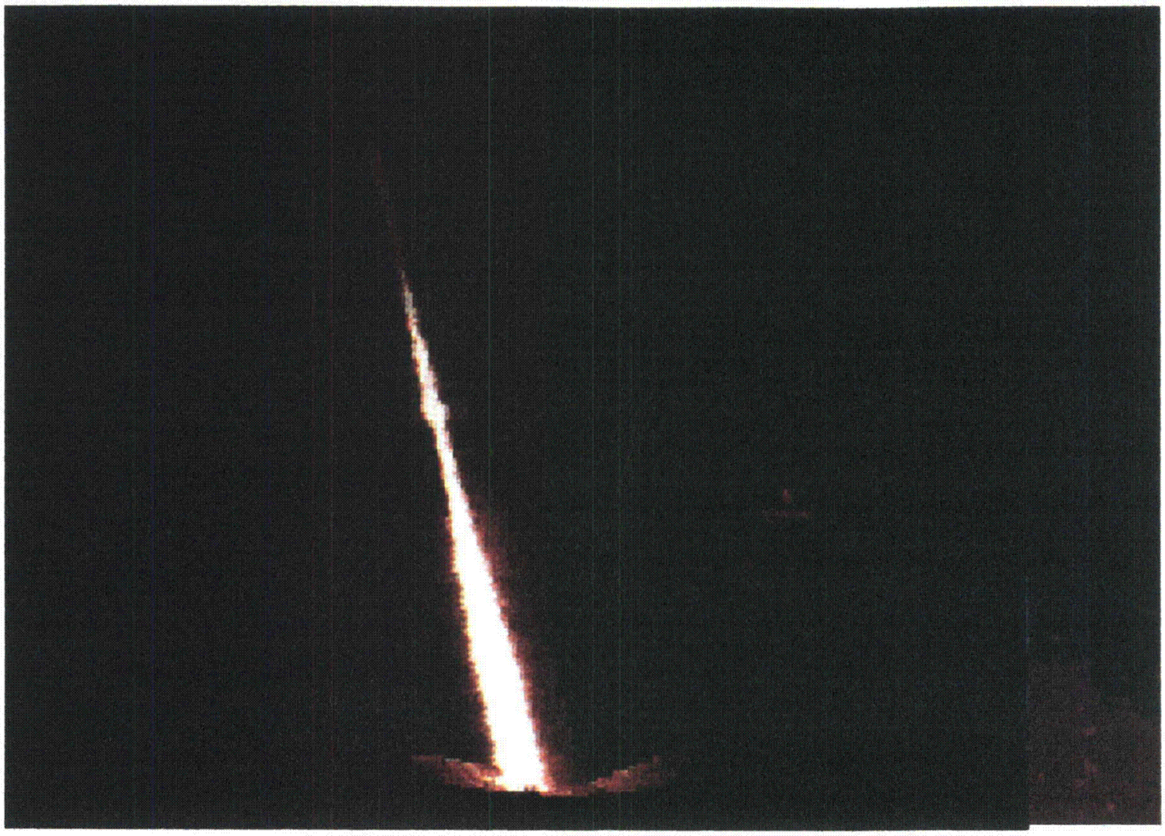
Hole 44-45

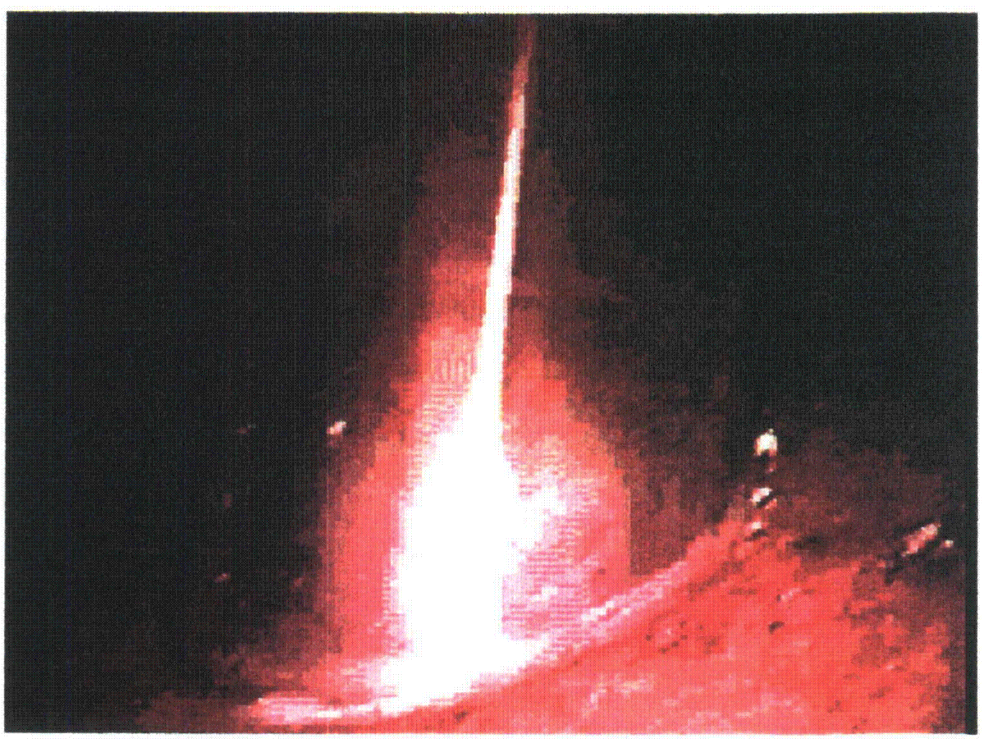




1996 Inspections

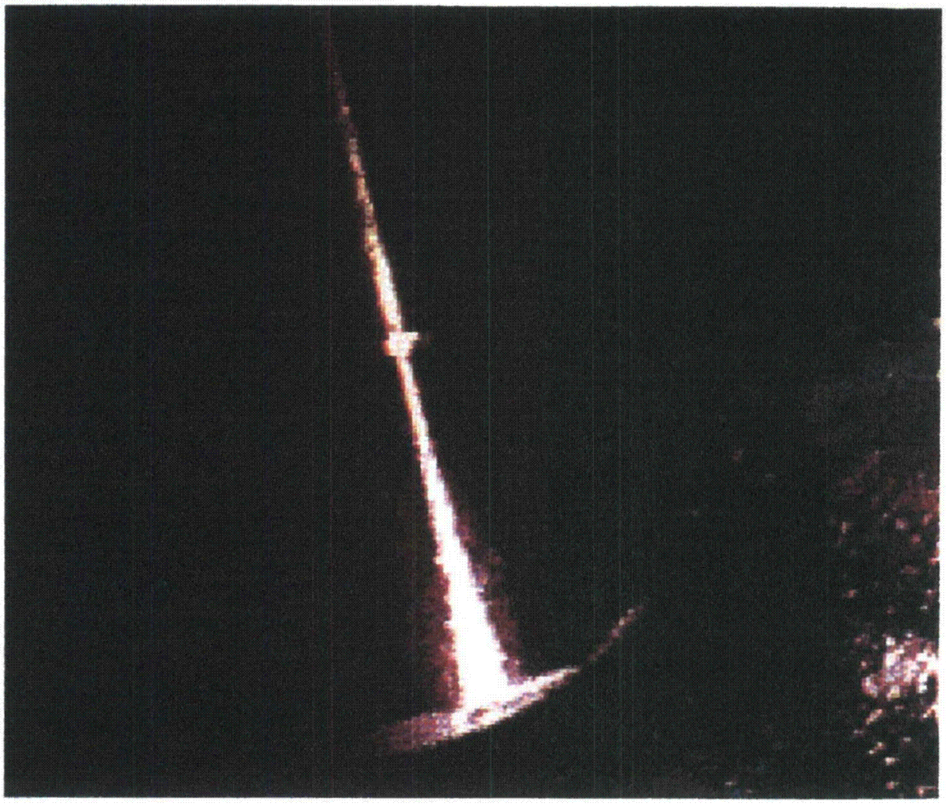
Hole 37-38

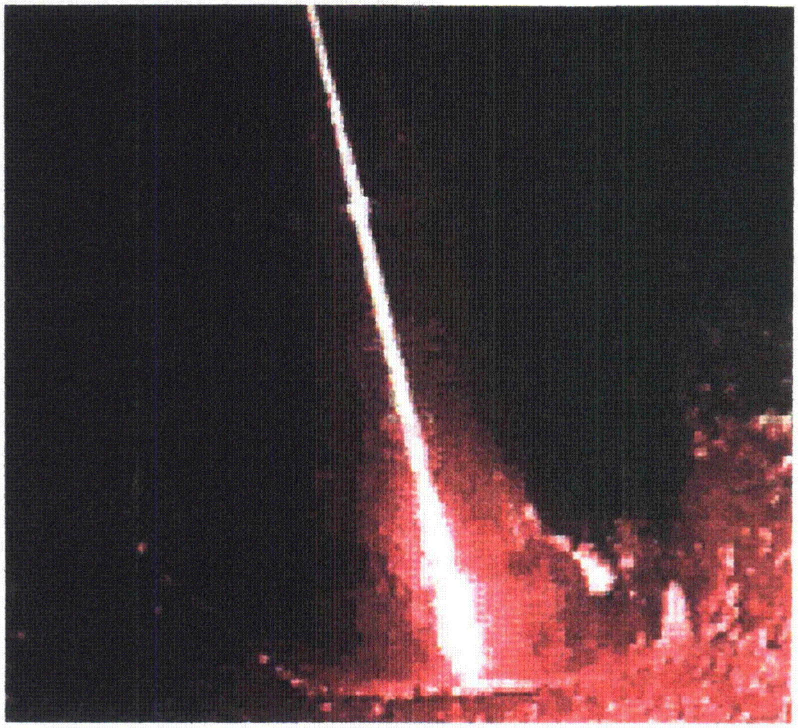


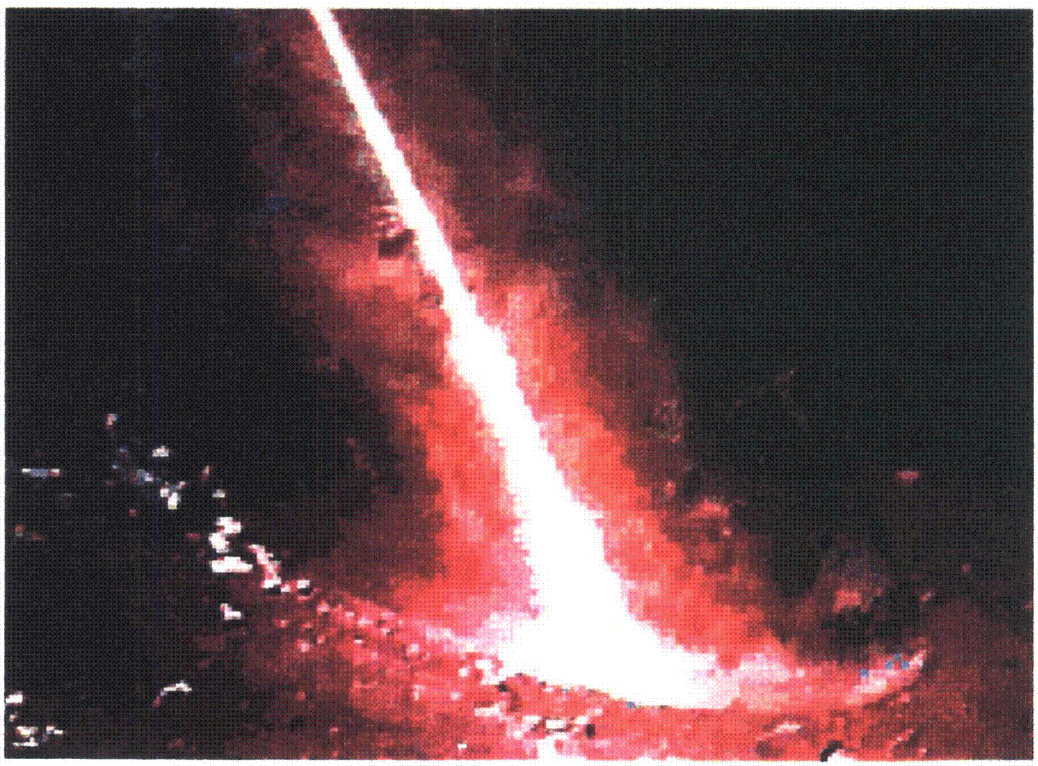


FENOC RESTRICTED INFORMATION

Hole 33-34



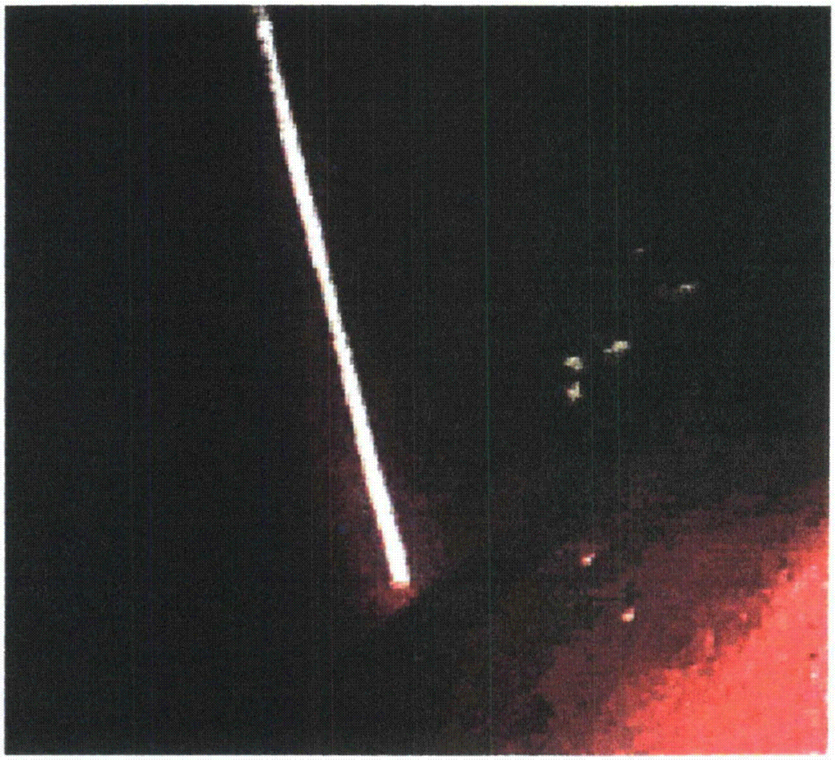




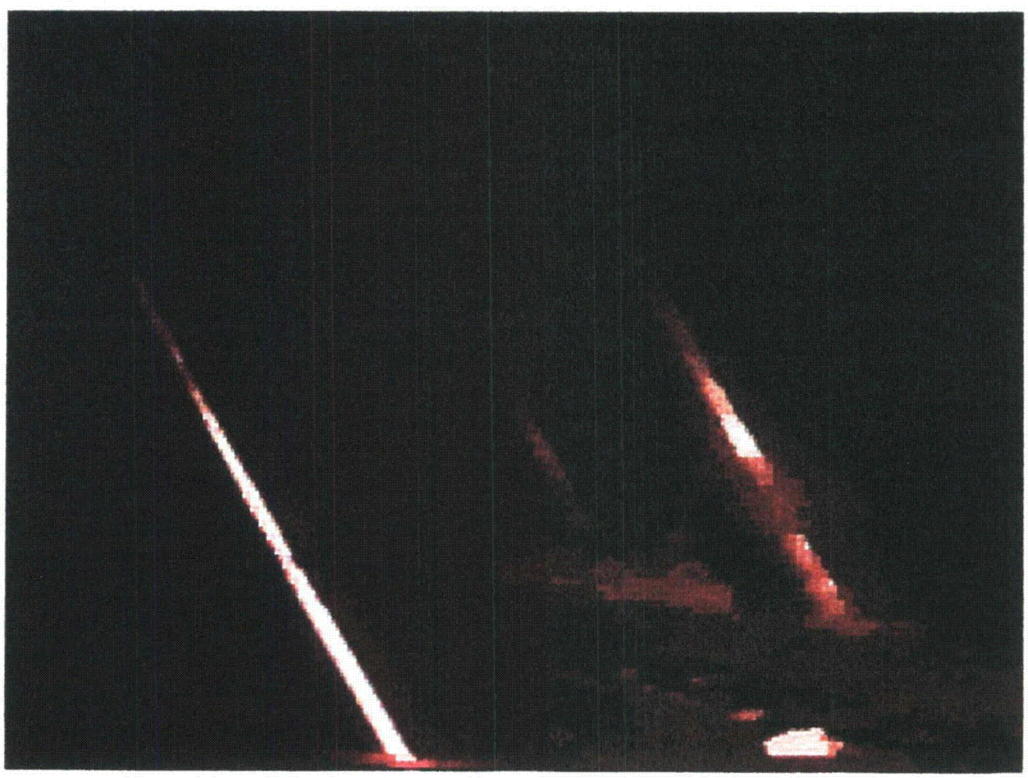
FENOC RESTRICTED INFORMATION

1996 Inspections

Hole 29-30

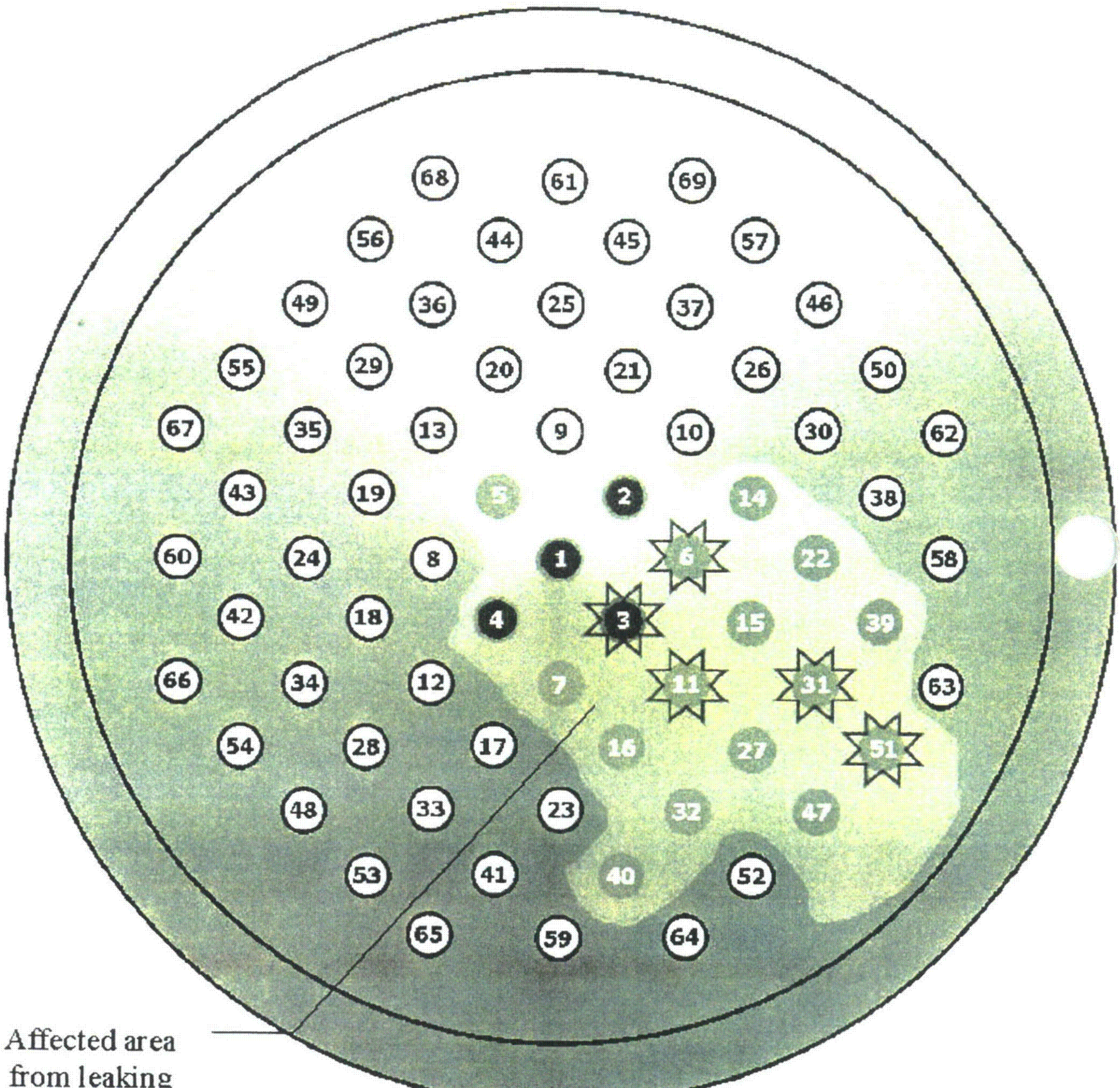


The boron deposits uphill of the CRDM drive below and to the right was reviewed from several angles and definite trails of born could be seen streaming from above the mirror insulation. This coupled with no boron on the bottom (downhill) edge of the CRDM penetration and the fact that boron will grow but not flow uphill allowed us to call this penetration as a non-leaker.



Spring 1998
Inspection

RPV Head 11 RFO Inspection Results



61 - No leakage identified

4 - Evaluated not to have sufficient gap to exhibit leakage

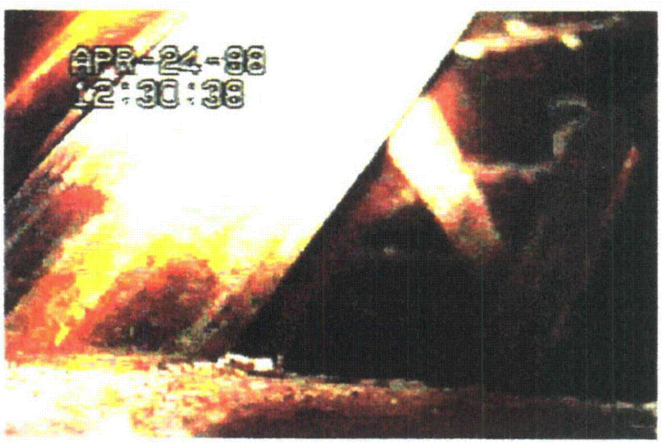
3 - Insufficient gap with leaking flange

5 - Nozzle obscured by boron

6 - Nozzle obscured by boron with leaking flange

No.53

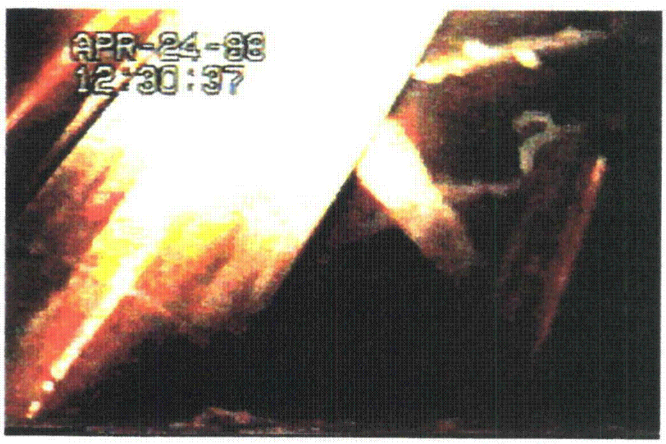
N0. 65



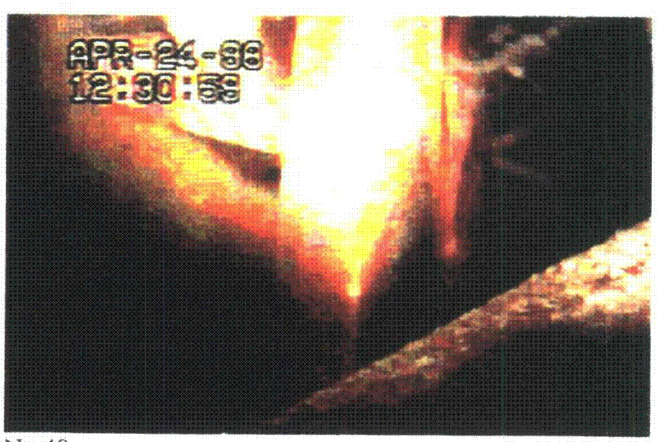
The following pictures are from access hole #9. They were clipped from video taken in the Spring of 1998. Although much more boron dusting was present in 1998 than in 1996, a good video inspection was able to be performed for those 50 drives that were not obscured by boron from leaking CRDM flanges. Although much more video can be viewed, these attached pictures are representative of the condition of the drives and the heads. We attempted to capture in still photographs all of the outer most drives since they are the most susceptible to circumferential cracking based upon finite element analysis which showed them to have the highest stresses on the uphill and downhill slopes of the penetration.

What can also be seen in many of the photos is the staining of the underside of the mirror insulation by boron trails. This corresponds to the boron found on top of the mirror insulation in the vicinity of the leaking CRDM flanges.

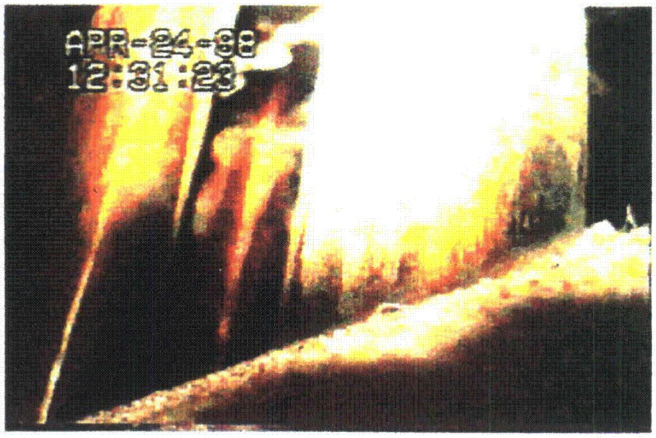
N0. 41



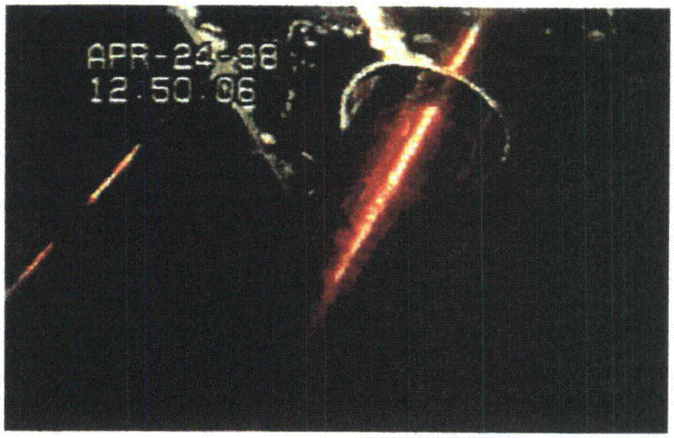
No. 33

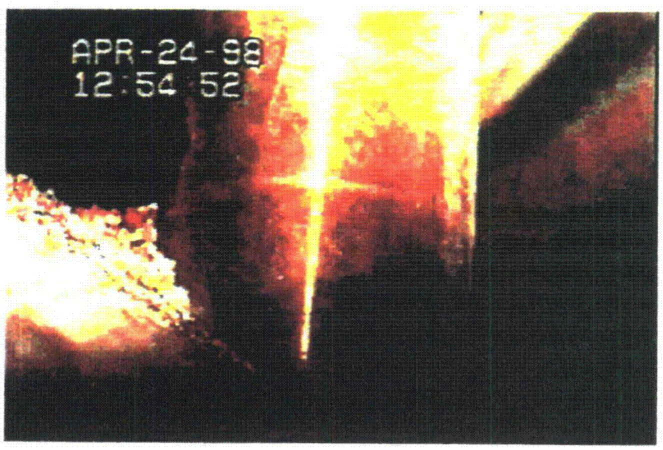


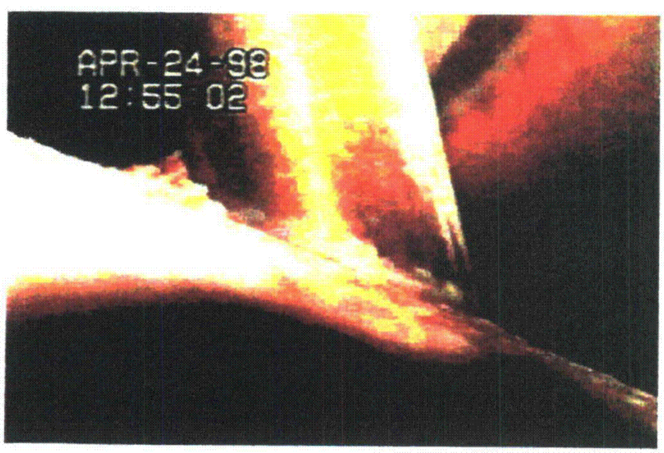
No.48



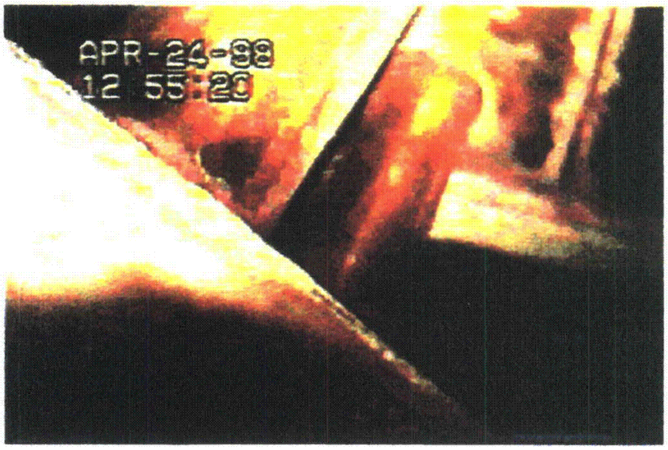
No. 65

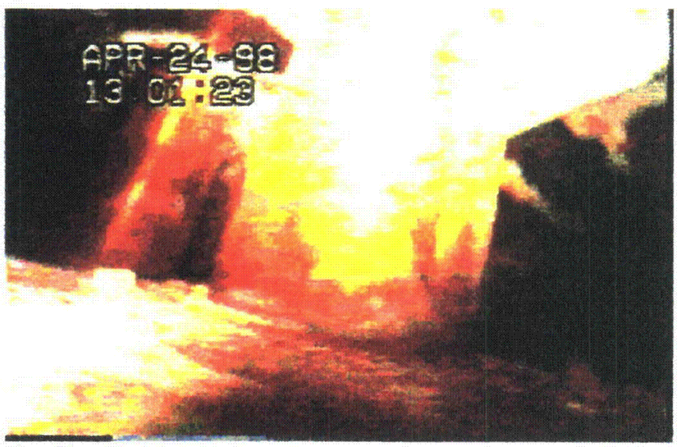






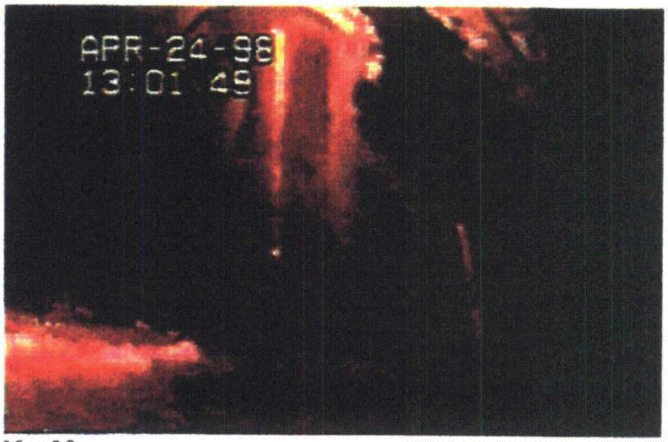
The two pictures to the left are examples of some drives where we had to view them from several angles to ascertain that the boron adjacent to the drives was actually boron that flowed or tumbled down from higher up on the head and came to rest against the uphill side of the CRDM nozzle. Sometimes this was ascertained by comparing the pictures at the left to video of the vacuuming that was performed later which showed the boron to very loose and not a crystalline mass. Additionally, there were no boron deposits on the downhill penetration seam, which is contrary, to what industry experience has shown us to be true at plants that have identified leakers. Because of the tight tolerances of the penetrations, any leakage through the penetration will encircle the drive with the largest accumulation being on the downhill edge because of gravity flow to that location.





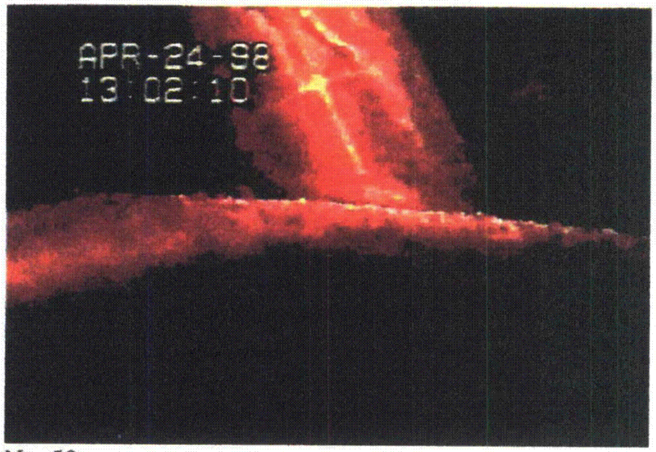
No. 69

No. 62

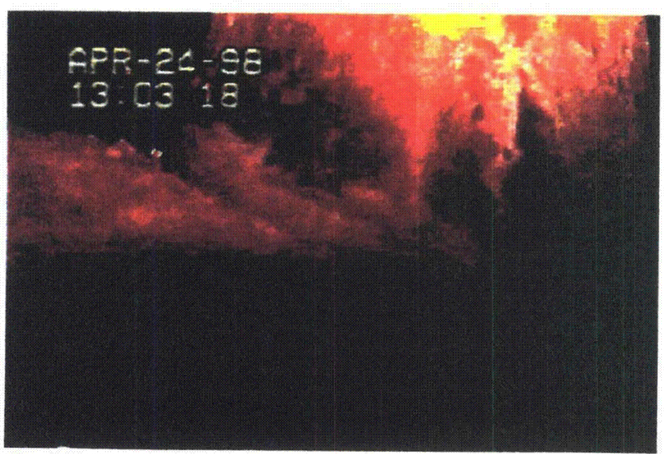


No. 38

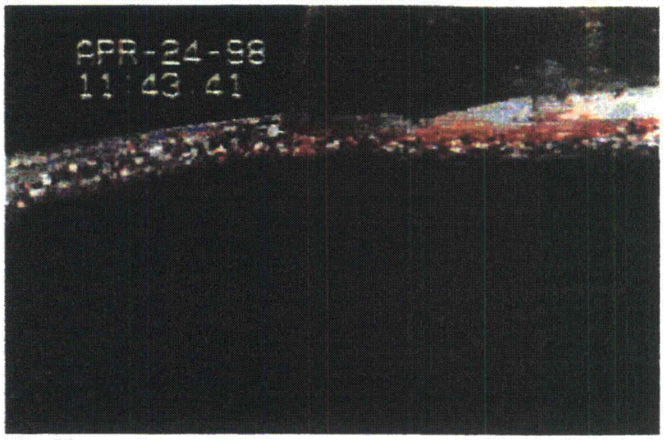
Note the loose boron clumps to the left which were not in the immediate vicinity of the nozzle penetrations. These clumps appeared to have accumulated further up on the head and then rolled or tumbled to their resting spots as shown. Note also the boron traces around the mirror insulation penetrations.



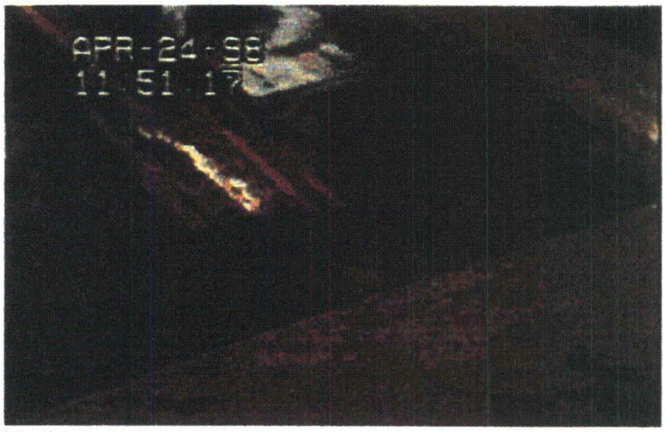
No. 50



No. 58

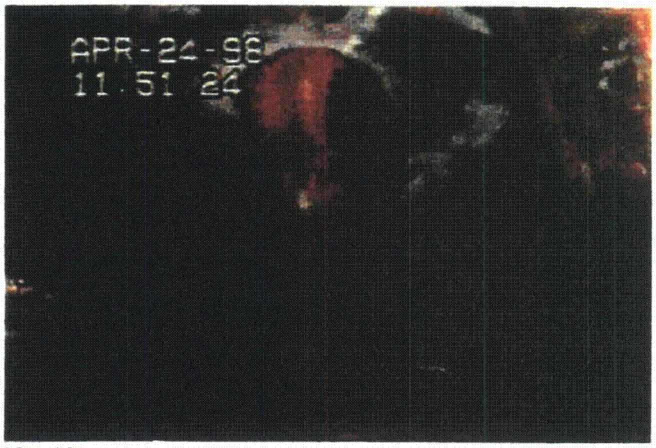


No. 63

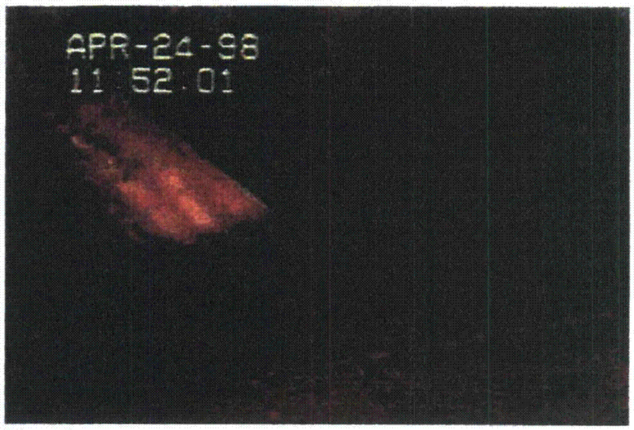


No.35

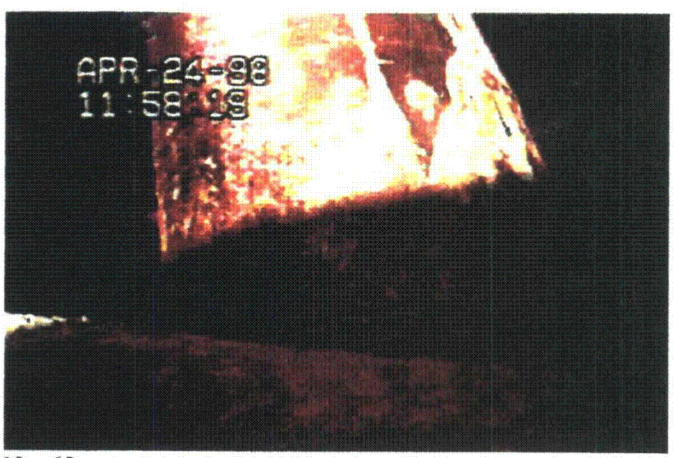
No. 42



No. 13

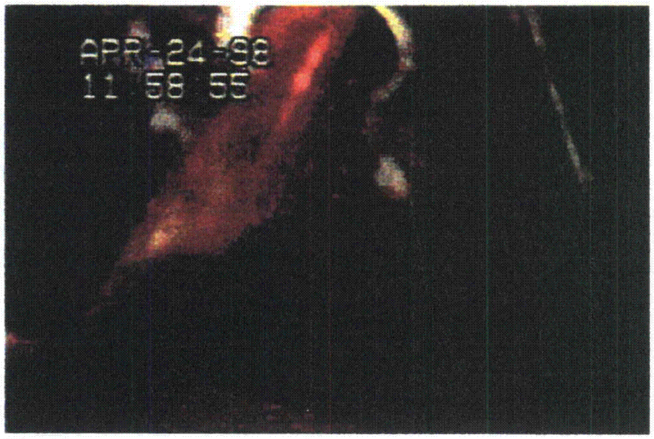


No. 43



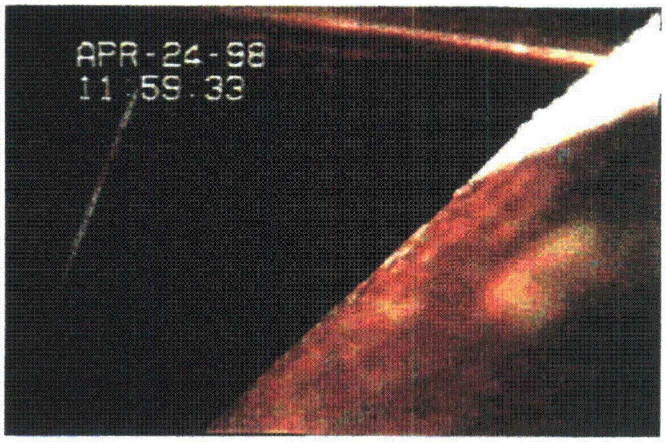
No. 60

No. 24

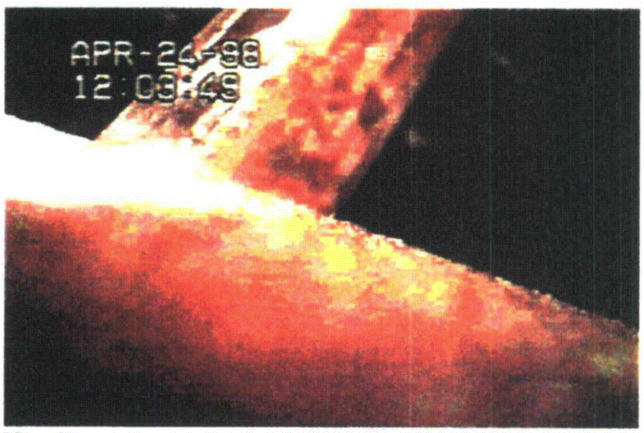


No.43

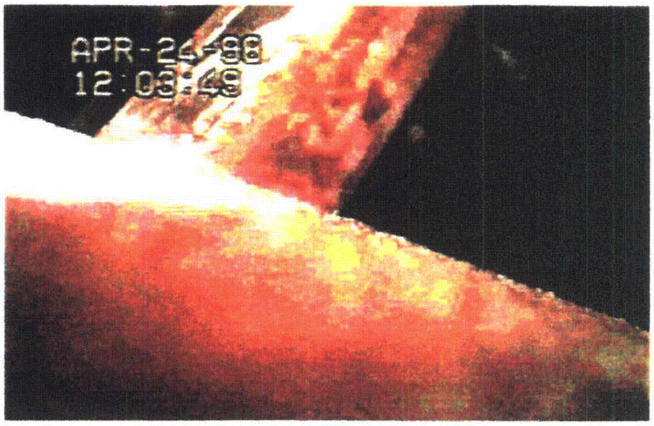
No. 67



No. 48, 54, 66

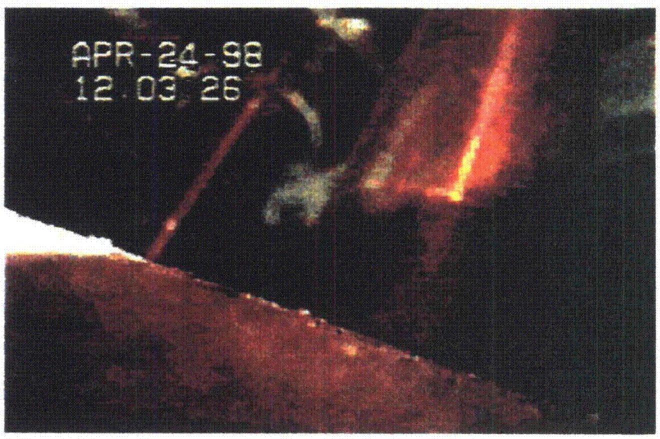


No. 67

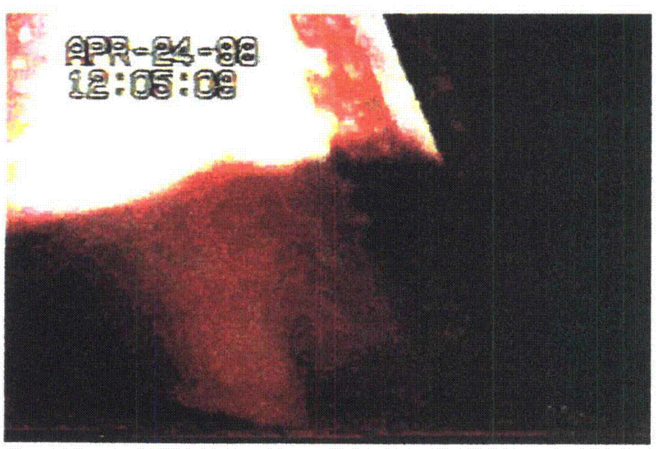


CRDM Penetrations as viewed from inspection opening #7

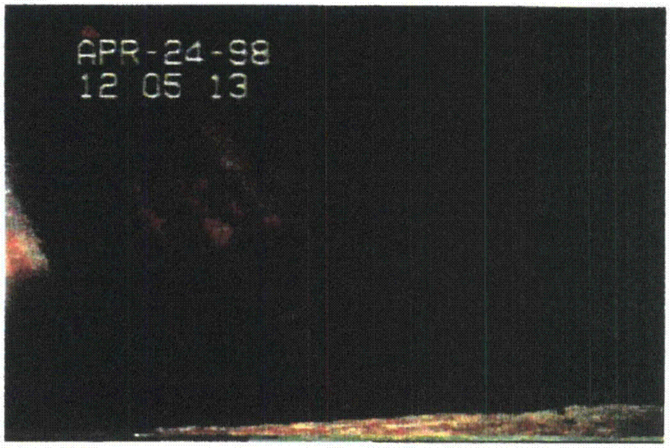
No. 56



No. 29



No49 side

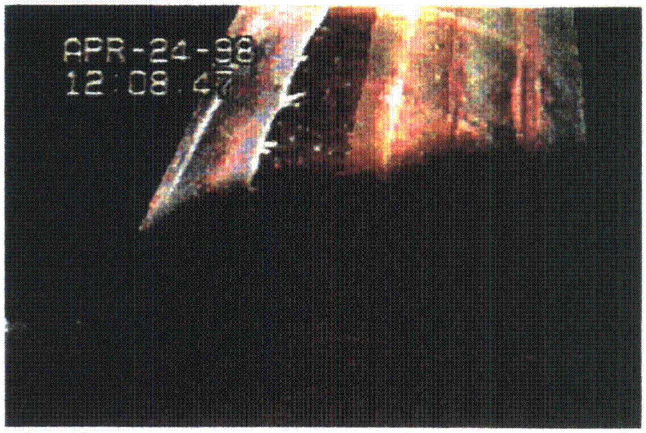


No. 55

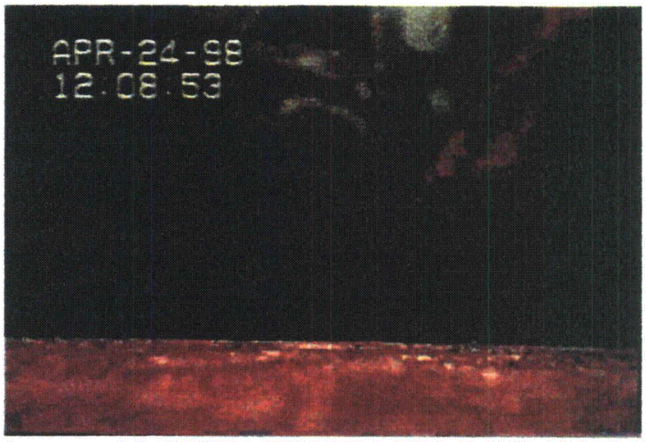
No. 49 front

No.36

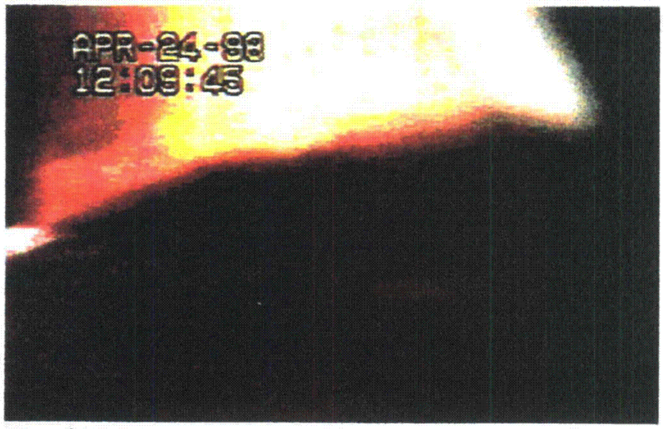
C26



No. 68

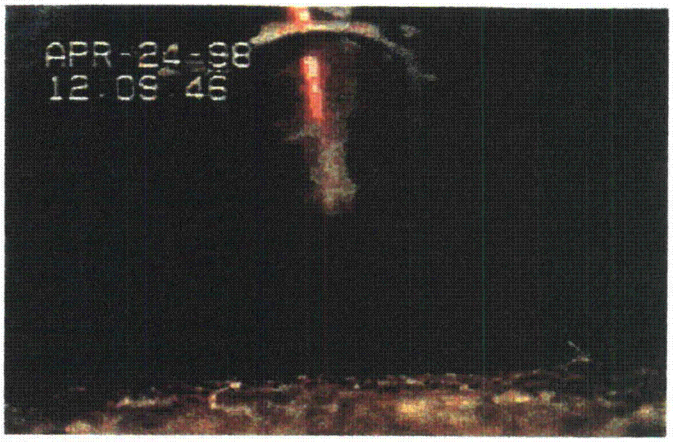


No. 44

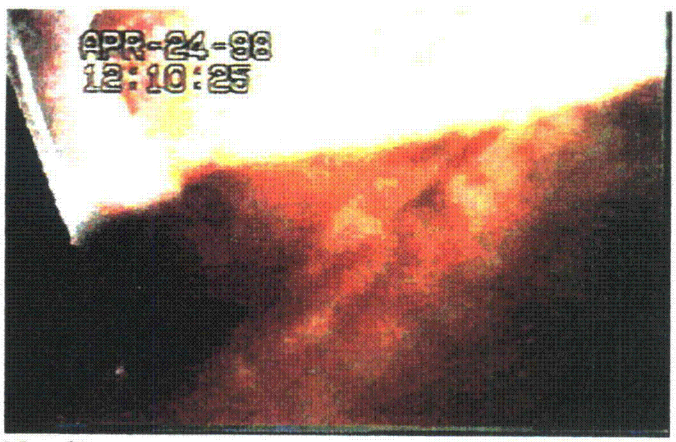


No. 61

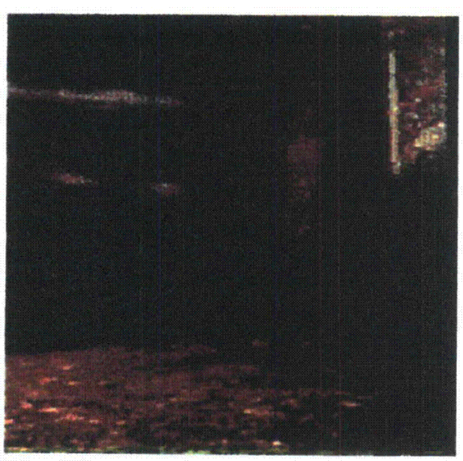
C27

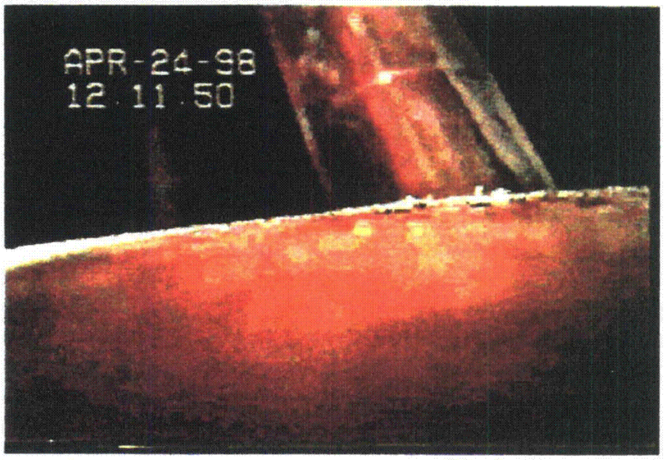


No. 25

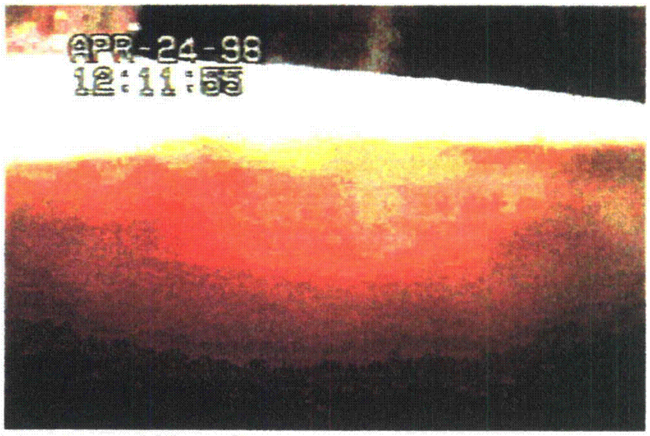


No. 61

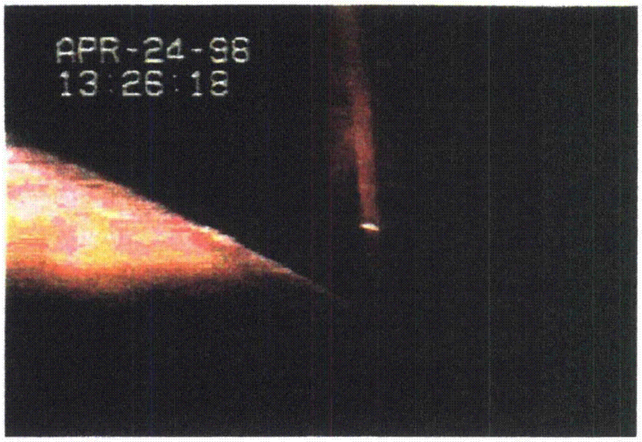

No. 25



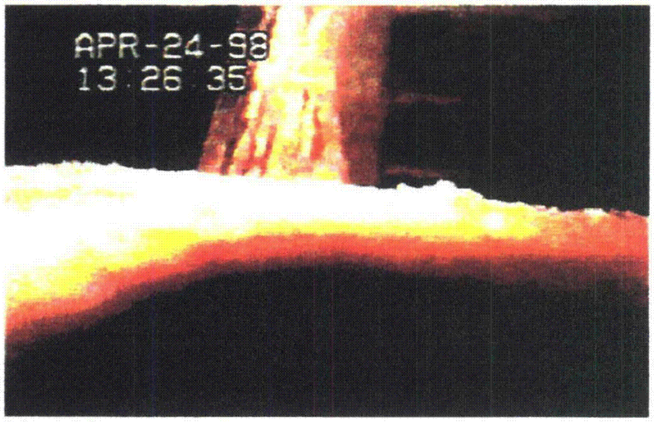
No. 68



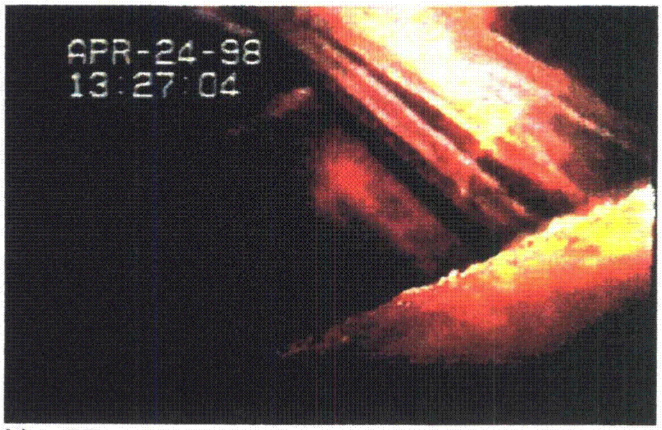
No. 69 and No. 45 in the middle on the back



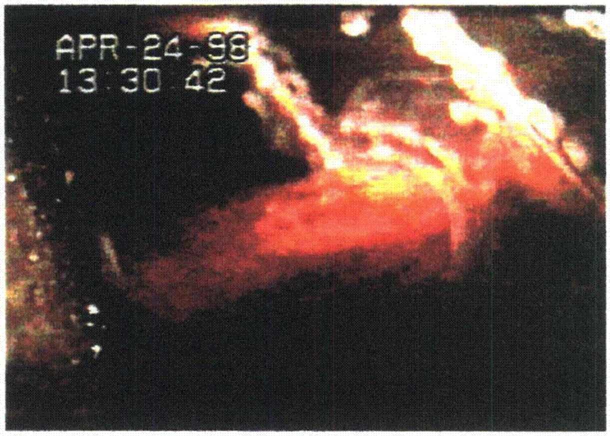
No. 57



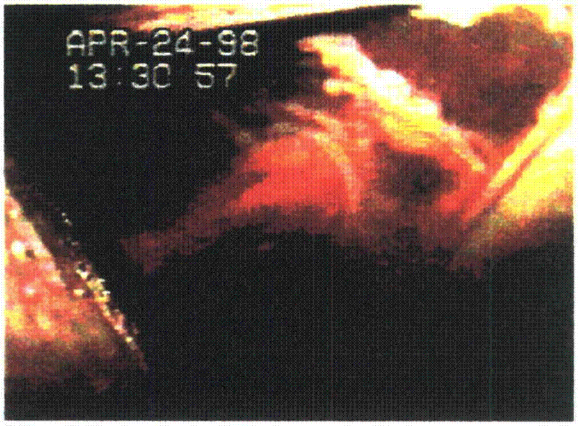
No. 46



No. 57



No. 37



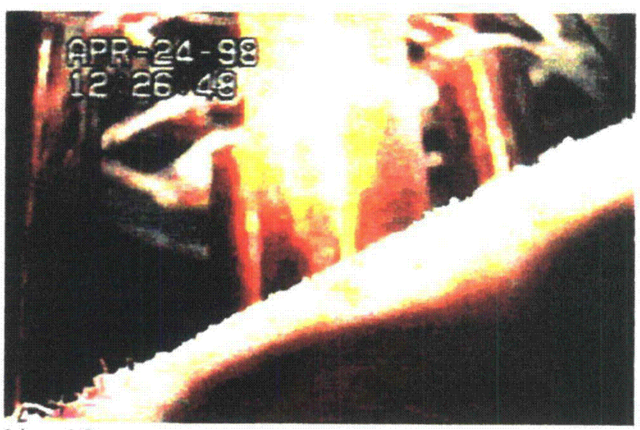
No. 26

No. 48

FENOC RESTRICTED INFORMATION

No. 34

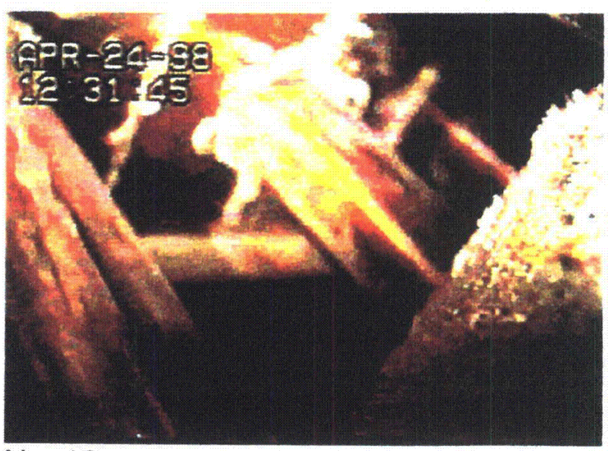
Same as above No. 34 on the right



No. 28

No. 48

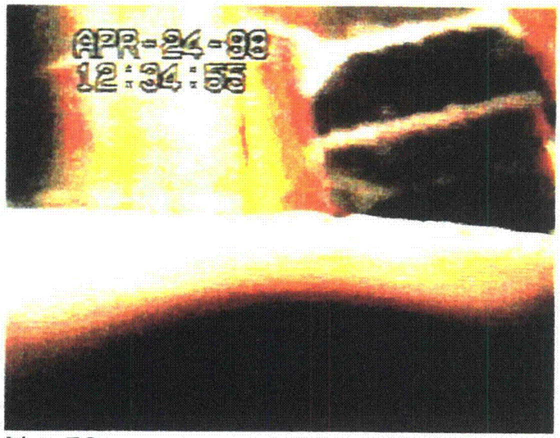
No. 66



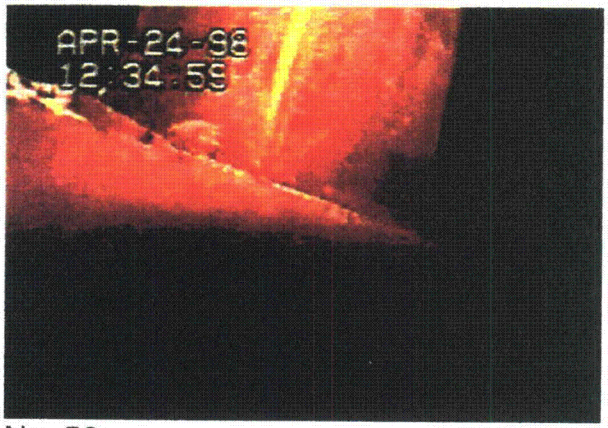
No. 18

FENOC RESTRICTED INFORMATION

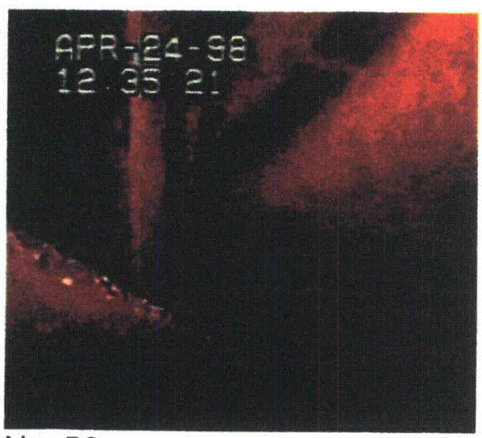
C33



No. 59



No. 59

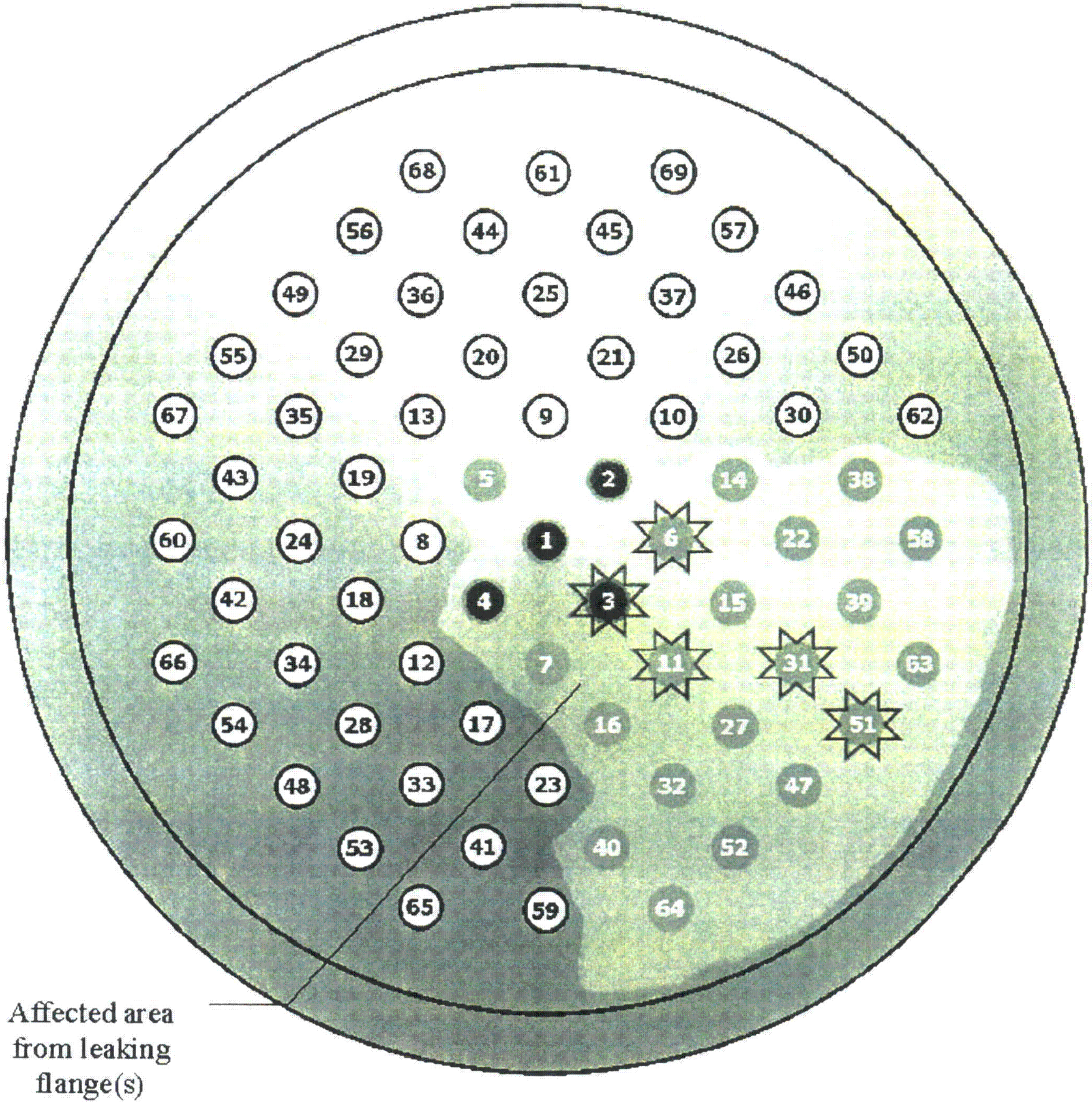


No. 52

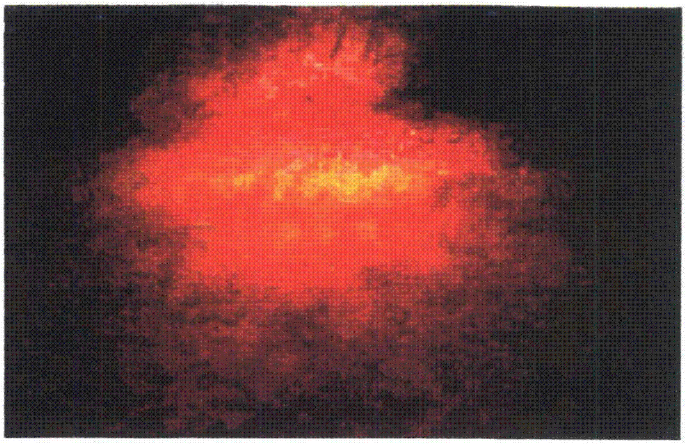
No. 59

Spring 2000 Inspection

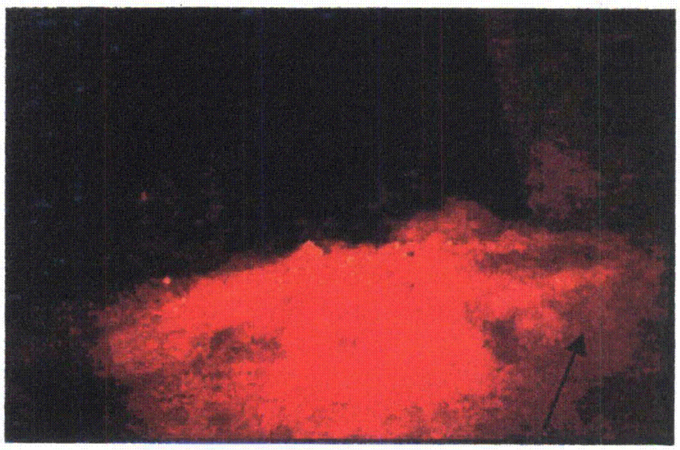
RPV Head 12 RFO Inspection Results



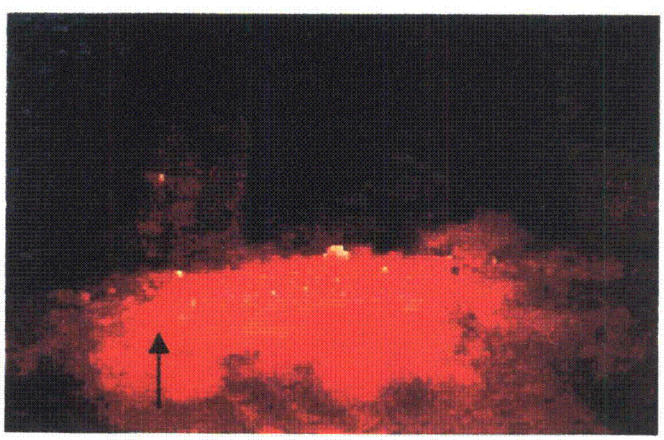
- No leakage identified
- Evaluated not to have sufficient gap to exhibit leakage
- Insufficient gap with leaking flange
- Nozzle obscured by boron
- Nozzle obscured by boron with leaking flange



No. 67



No. 43



No. 35

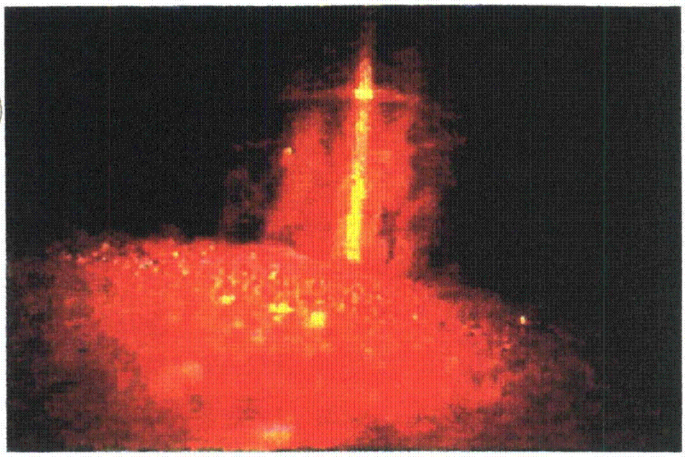
These photos were taken from our 2000 spring outage videotapes.

The lighting and video camera optics created an orange coloration of all of the pictures. However, deposits of boron are visually discernable as shown by the scattered pieces of boron.

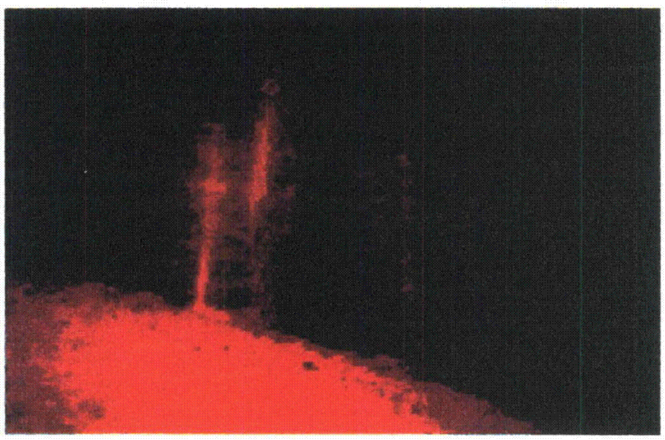
No 67 has no buildup around its penetration and the boron debris shown in the picture for No. 43 are scattered well away from the penetration.

These drives were video taped because they had boron deposits in the vicinity of the CRDMs. Completely clean drive penetrations are not depicted here.

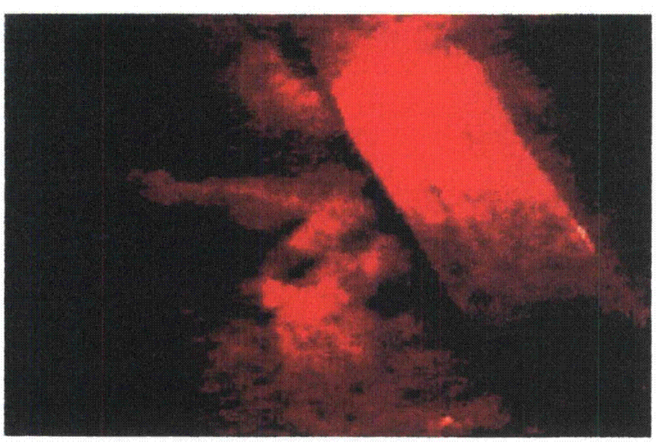
The photo for No. 19 depicts in the background the extent of boron buildup on the head and is the reason no credit is taken for being able to visually inspect the remainder of the drives.



No. 60

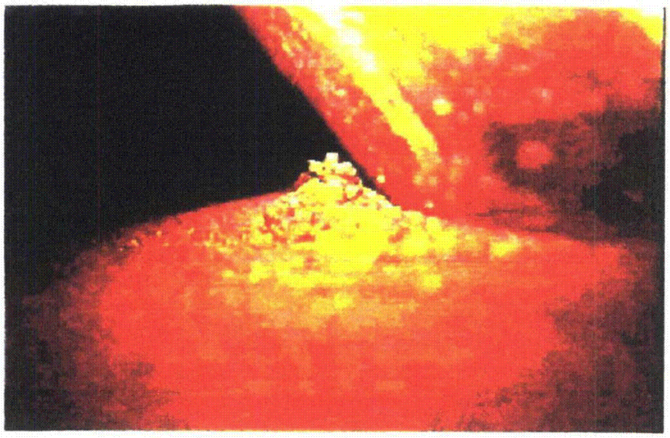


No. 24

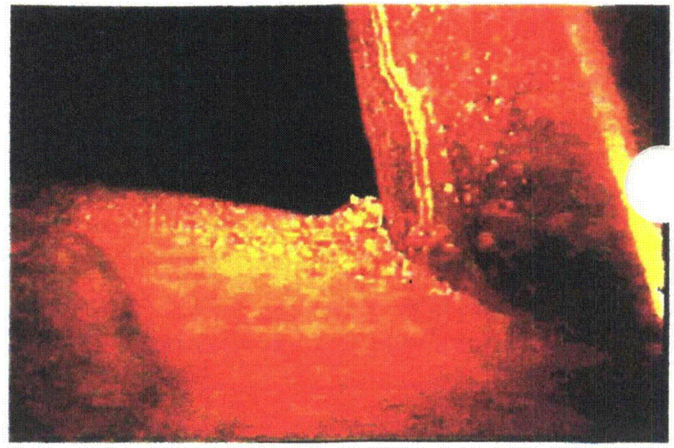


No. 19

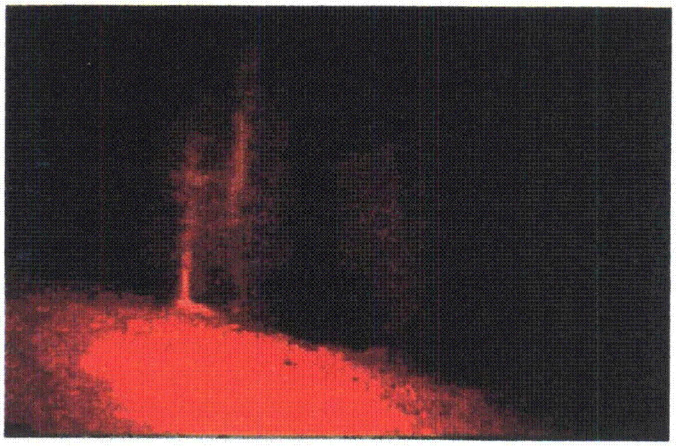
The debris piled up against the uphill side of No. 66 on the next page is indicative of loose debris that has fallen down the slope of the head and came to rest on the drive. It does not resemble "popcorn" deposits witnessed at other plants. There were also no signs of boron anywhere else on the drive penetration opening.



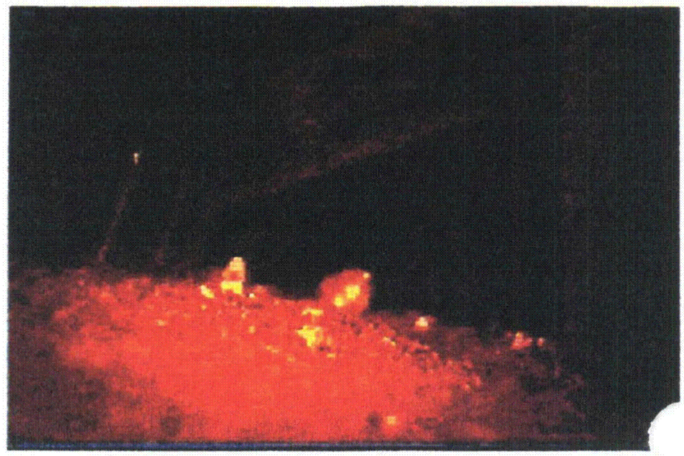
No. 66



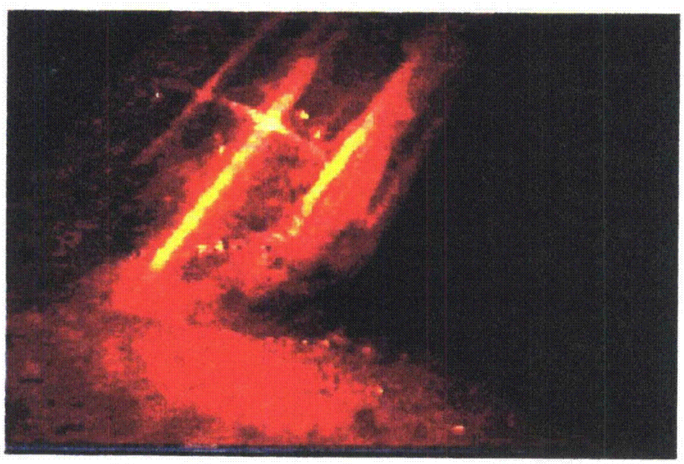
No. 66



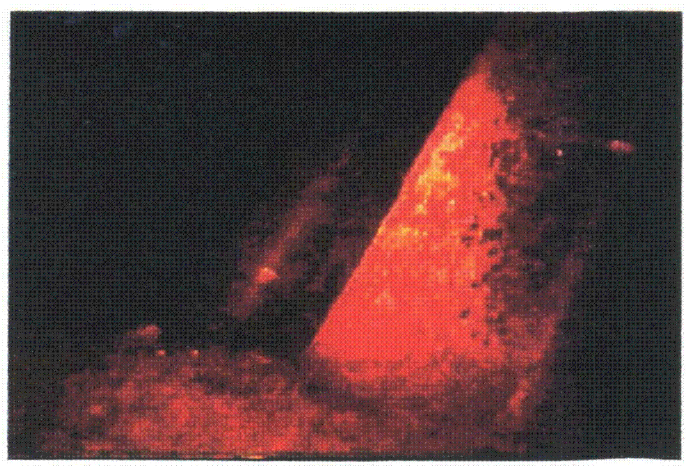
No. 42



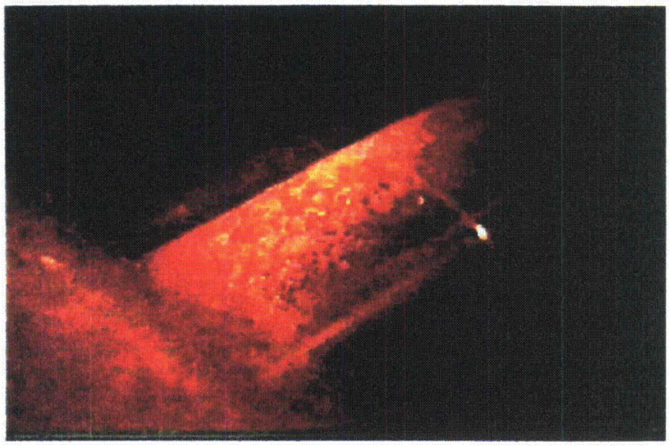
No. 19 No. 24



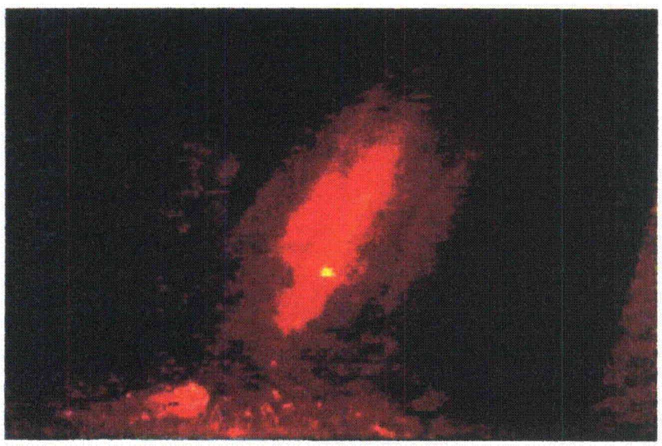
No. 35



No. 35



No. 55



No. 29

10 CFR 2.790 Affidavit

(2 pages follow)

AFFIDAVIT OF STEVEN P. MOFFITT

A. My name is Steven P. Moffitt. I am Director – Technical Services for FirstEnergy Nuclear Operating Company ("FENOC") at the Davis-Besse Nuclear Power Station, Unit 1 ("DBNPS-1"), and as such, I am authorized to execute this Affidavit.

B. I am familiar with the criteria applied by FENOC to determine whether certain FENOC information is proprietary and I am familiar with the procedures established with FENOC to ensure the proper application of these criteria.

C. I am familiar with the FENOC information included in the DBNPS-1 Tenth Refueling Outage, Eleventh Refueling Outage, and Twelfth Refueling Outage Reactor Vessel Head Inspection photographs and hereto referred to as "Photographs". Information contained in these Photographs has been classified by FENOC as **Restricted** in accordance with the policies established by FENOC for the control and protection of confidential and proprietary information.

D. These Photographs are being made available to the U.S. Nuclear Regulatory Commission in confidence with a statement that it is **Restricted** information and a request that the information contained in these Photographs be withheld from public disclosure.

E. The following information is provided to demonstrate that the provisions of the Code of Federal Regulations, Title 10 – Energy, Part 2, Section 790 have been considered in the confidential and commercial classification of these Photographs as **Restricted**:

(i) These Photographs have been held in confidence by FENOC. Copies of these Photographs are clearly marked as **Restricted**.

(ii) These Photographs contain information of a proprietary and confidential nature and is of the type customarily held in confidence by FENOC and not made available to the public.

(iii) These Photographs are being transmitted to the U.S. Nuclear Regulatory Commission in confidence.

(iv) These Photographs are not available in public sources.

(v) These Photographs contain confidential and commercial information regarding the material condition of certain components of the DBNPS-1 that can be subject to future negotiated commercial purchase agreements with a vendor(s) external to FENOC. The information provided on these Photographs is of a nature that cannot be acquired or duplicated by others.

F. In accordance with FENOC's policies governing the protection and control of information, **Restricted** information contained in these Photographs has been made available, on a limited basis, outside FENOC only as required and under suitable non- disclosure agreement providing limited use of the information.

G. FENOC requires that **Restricted** information contained in these Photographs be kept in a secured file or area and distributed only on a need-to-know basis.

H. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

By: Steven P. Moffitt
Steven P. Moffitt

Affirmed and subscribed before me this 30th Day of October, 2001.

Laura A. Jennison
Notary Public, State of Ohio
My Commission Expires August 16, 2006

Commitment List

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions the DBNPS. They are described only for information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs (419-321-8450) at the DBNPS of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
None	

NRC LETTERS - REVIEW AND APPROVAL REPORT
ED 7159-7

RAS C-155

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 14
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

(1) RECORD

(3) SUMMARY (Log No., Title Subject)
Transmittal of Results of RVP Head CRDM Nozzle Visual Examinations

(4) COMMITMENT LIST ADDED TO LETTER ☒

(5) PERIODIC
☐ YES

(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC
Target Date 11/01/01 ☐ N/A

(7) SPECIAL
☒ EXPRE

(8) PREPARED BY
Rod Cook ext. 7782

(9) NOTARY
☒ YES

(11) ADDITIONAL REFERENCES
Serial 2731
Serial 2735

(12) COMMITMENT NO.(S) CLOSED

DOCKETED
USNRC
September 9, 2009 (11:00am)
OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

(13) COMMENTS

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ COGNIZANT REGULATORY AFFAIRS INDIVIDUAL	R.M. Cook	[signature]	10/30/2001	10/30/2001
☒ MANAGER, DESIGN ENGINEERING	D. Geisen	[signature]	10/30/2001	10/30/2001
☒ DIRECTOR, TECHNICAL SERVICES	S. Moffitt	See affidavit		
☐				
☐				
☐				
☐				
☐				
☒ SUPERVISOR, DB LICENSING	D.R. Wuokko	DRW	10/30/01	10/30/01
☒ SUPERVISOR, DB COMPLIANCE	D.L. Miller	DLM	10/30/01	10/30/01
☒ MANAGER, REGULATORY AFFAIRS	D. H. Lockwood	DHL	10-30-01	10-30-01
☒ VICE PRESIDENT	G.G. Campbell	[signature]	OCT 30 2001	10/30/01

DATE ADDED TO LETTER ☑
10-30-01

(15) DATE ADDED BY
Laura A. Jennison

DATE SENT TO NRC
10-31-01

(16) DISTRIBUTED BY
Laura A. Jennison

DATE OF BLIND DISTRIBUTION
11-6-01

(18) DISTRIBUTED BY
Laura A. Jennison

(17) ADDITIONAL DISTRIBUTION

S11-00602

TEMPLATE = SECY-028 DS 02

NRC027-1702

NRC LETTERS - REVIEW AND APPROVAL REPORT
ED 7159-7

(1) RECORDS MANAGEMENT NO. RAS-01-00479	(2) SERIAL NO. 2744

(3) SUMMARY (Log No., Title Subject)
Transmittal of Results of RVP Head CRDM Nozzle Visual Examinations

(4) COMMITMENT LIST ADDED TO LETTER ☒	(5) PERIODIC / NON-PERIODIC REPORT ☐ YES ☒ NO REPORT NO.	
(6) DATE RESPONSE DUE TO BE SUBMITTED TO NRC Target Date 11/01/01 ☐ N/A	7) SPECIAL HANDLING PICKETT + SANDS ☒ EXPRESS DELIVERY ☐ TELECOPY	DATE SENT 10-30-01
(8) PREPARED BY Rod Cook ext. 7782	(9) NOTARY ☒ YES ☐ NO	(10) LICENSE FEE REQUIRED ☐ YES ☒ NO

(11) ADDITIONAL REFERENCES

Serial 2731
Serial 2735

(12) COMMITMENT NO.(S) CLOSED

(13) COMMENTS

(14) REVIEW AND APPROVAL		INITIALS	DATE RECEIVED	DATE APPROVED
☒ COGNIZANT REGULATORY AFFAIRS INDIVIDUAL	R.M. Cook	[signature]	10/30/2001	10/30/2001
☒ MANAGER, DESIGN ENGINEERING	D. Geisen	[signature]	10/30/2001	10/30/2001
☒ DIRECTOR, TECHNICAL SERVICES	S. Moffitt	See affidavit		
☐				
☐				
☐				
☐				
☐				
☒ SUPERVISOR, DB LICENSING	D.R. Wuokko	DRW	10/30/01	10/30/01
☒ SUPERVISOR, DB COMPLIANCE	D.L. Miller	DLM	10/30/01	10/30/01
☒ MANAGER, REGULATORY AFFAIRS	D. H. Lockwood	DHL	10-30-01	10-30-01
☒ VICE PRESIDENT	G.G. Campbell	[signature]	OCT 30 2001	10/30/01

DATE ADDED TO LETTER ☑ 10-30-01	(15) DATE ADDED BY Laura A. Dennison	(17) ADDITIONAL DISTRIBUTION
DATE SENT TO NRC 10-31-01	(16) DISTRIBUTED BY Laura A. Dennison	
DATE OF BLIND DISTRIBUTION 11-6-01	(18) DISTRIBUTED BY Laura A. Dennison	

S11-00602

NRC027-1702

The NRC Letters - Review and Approval Report (ED 7159-7) should be completed by the Regulatory Affairs Section.

BLOCK 1	RECORDS MANAGEMENT NO. - Regulatory Affairs enters Records Management number prior to distribution of correspondence to NRC.
BLOCK 2	SERIAL NO. - Initiator enters serial number obtained from the Regulatory Affairs Clerk.
BLOCK 3	SUMMARY (Log No., Title Subject) - Initiator enters a summary of the correspondence. This summary should identify if the correspondence is in response to any previous correspondence and why the letter is being written.
BLOCK 4	COMMITMENT LIST ADDED TO LETTER - Preparers checks the block to indiate a commitment list has been included with the letter.
BLOCK 5	PERIODIC/NON-PERIODIC REPORT - Identify whether this correspondence is a Periodic or Non-Periodic Report as identified in Nuclear Group Procedure NG-NS-00807.
BLOCK 6	DATE RESPONSE DUE TO BE SUBMITTED TO NRC - Initiator enters the date the correspondence is due to the NRC. If the correspondence does not have a required due date, the block shall be marked not applicable (NA).
BLOCK 7	SPECIAL HANDLING - Initiator checks if the correspondence requires special distribution to the NRC. If yes, the Regulatory Affairs clerk enters date the correspondence is sent.
BLOCK 8	PREPARED BY - Initiator enters the names of individuals responsible for providing technical information for the correspondence along with his/her name.
BLOCK 9	NOTARY - Initiator checks if the correspondence is required to be notarized.
BLOCK 10	LICENSE FEE REQUIRED - Initiator checks if a license fee is required, per the requirements of 10 Part 170. If yes, the Initiator shall complete a Voucher Check Authorization (Form 294) and obtain the appropriate fees to accompany the correspondence.
BLOCK 11	ADDITIONAL REFERENCES - Initiator enters any additional NRC correspondence or documents that pertain to the subject correspondence.
BLOCK 12	COMMITMENT NO(S). CLOSED - Initiator enters the Commitment Management System number(s) of any commitments that are closed by the subject correspondence.
BLOCK 13	COMMENTS - Initiator or any reviewer enters appropriate comments regarding the subject correspondence.
BLOCK 14	REVIEW AND APPROVAL - Initiator checks and /or enters the desired reviewer(s) . The technical accuracy of a response to the NRC is the responsibility of the Director and Management individual assigned the action.
BLOCK 15	DATE ADDED BY - Distributor checks the Date Added to Letter Block and signs the Date Added By block to indicate the original letter has been dated prior to distribution to the NRC.
BLOCK 16	DISTRIBUTED BY - Distribution to the NRC shall be made by the Regulatory Affairs Section. Distributor signs the Distributed By block and completes the Date Sent to NRC block.
BLOCK 17	ADDITIONAL DISTRIBUTION - Initiator enters individuals requiring distribution that are not on the standard distribution list.
BLOCK 18	DISTRIBUTED BY - Distributor signs the Distributed By block and completes the Date of Blind Distribution block.

S11-00603

RAS C-156

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 15
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

§ 50.55(f)(3), or a change to a licensee's NRC-accepted quality assurance topical report under § 50.54(a)(3) or § 50.55(f)(3), must be submitted to the NRC's Document Control Desk, with a copy to the appropriate Regional Office, and a copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility. If the communication is on paper, the submission to the Document Control Desk must be the signed original.

(ii) A change to an NRC-accepted quality assurance topical report from nonlicensees (*i.e.*, architect/engineers, NSSS suppliers, fuel suppliers, constructors, etc.) must be submitted to the NRC's Document Control Desk. If the communication is on paper, the signed original must be sent.

(8) *Certification of permanent cessation of operations.* The licensee's certification of permanent cessation of operations, under § 50.82(a)(1), must state the date on which operations have ceased or will cease, and must be submitted to the NRC's Document Control Desk. This submission must be under oath or affirmation.

(9) *Certification of permanent fuel removal.* The licensee's certification of permanent fuel removal, under § 50.82(a)(1), must state the date on which the fuel was removed from the reactor vessel and the disposition of the fuel, and must be submitted to the NRC's Document Control Desk. This submission must be under oath or affirmation.

(c) *Form of communications.* All paper copies submitted to meet the requirements set forth in paragraph (b) of this section must be typewritten, printed or otherwise reproduced in permanent form on unglazed paper. Exceptions to these requirements imposed on paper submissions may be granted for the submission of micrographic, photographic, or similar forms.

(d) *Regulation governing submission.* Licensees and applicants submitting correspondence, reports, and other written communications under the regulations of this part are requested but not required to cite whenever practical, in the upper right corner of the first page of the submission, the specific regulation or other basis requiring submission.

(e) *Conflicting requirements.* The communications requirements contained in this section and §§ 50.12, 50.30, 50.36, 50.36a, 50.44, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.62, 50.71, 50.73, 50.82, 50.90, and 50.91 supersede and replace all existing requirements in any license conditions or technical specifications in effect on January 5, 1987. Exceptions to these requirements must be approved by the Office of Information Services, Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone (301) 415-7233, e-mail *INFOCOLLECTS@nrc.gov*.

[68 FR 58808, Oct. 10, 2003]

§ 50.5 Deliberate misconduct.

(a) Any licensee, applicant for a license, employee of a licensee or applicant; or any contractor (including a supplier or consultant), subcontractor, employee of a contractor or subcontractor of any licensee or applicant for a license, who knowingly provides to any licensee, applicant, contractor, or subcontractor, any components, equipment, materials, or other goods or services that relate to a licensee's or applicant's activities in this part, may not:

(1) Engage in deliberate misconduct that causes or would have caused, if not detected, a licensee or applicant to be in violation of any rule, regulation, or order; or any term, condition, or limitation of any license issued by the Commission; or

(2) Deliberately submit to the NRC, a licensee, an applicant, or a licensee's or applicant's contractor or subcontractor, information that the person submitting the information knows to be incomplete or inaccurate in some respect material to the NRC.

(b) A person who violates paragraph (a)(1) or (a)(2) of this section may be subject to enforcement action in accordance with the procedures in 10 CFR part 2, subpart B.

(c) For the purposes of paragraph (a)(1) of this section, deliberate misconduct by a person means an intentional act or omission that the person knows:

(1) Would cause a licensee or applicant to be in violation of any rule, regulation, or order; or any term, condition, or limitation, of any license issued by the Commission; or

TEMPLATE = SECY-028

DS 02

(2) Constitutes a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee, applicant, contractor, or subcontractor.

[63 FR 1897, Jan. 13, 1998]

§50.7 Employee protection.

(a) Discrimination by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant against an employee for engaging in certain protected activities is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, or privileges of employment. The protected activities are established in section 211 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act or the Energy Reorganization Act.

(1) The protected activities include but are not limited to:

(i) Providing the Commission or his or her employer information about alleged violations of either of the statutes named in paragraph (a) introductory text of this section or possible violations of requirements imposed under either of those statutes;

(ii) Refusing to engage in any practice made unlawful under either of the statutes named in paragraph (a) introductory text or under these requirements if the employee has identified the alleged illegality to the employer;

(iii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements;

(iv) Testifying in any Commission proceeding, or before Congress, or at any Federal or State proceeding regarding any provision (or proposed provision) of either of the statutes named in paragraph (a) introductory text.

(v) Assisting or participating in, or is about to assist or participate in, these activities.

(2) These activities are protected even if no formal proceeding is actually initiated as a result of the employee assistance or participation.

(3) This section has no application to any employee alleging discrimination prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.

(b) Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 180 days after an alleged violation occurs. The employee may do this by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.

(c) A violation of paragraph (a), (e), or (f) of this section by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant may be grounds for—

(1) Denial, revocation, or suspension of the license.

(2) Imposition of a civil penalty on the licensee, applicant, or a contractor or subcontractor of the licensee or applicant.

(3) Other enforcement action.

(d) Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.

(e)(1) Each licensee and each applicant for a license shall prominently post the revision of NRC Form 3, "Notice to Employees," referenced in 10 CFR 19.11(c). This form must be posted at locations sufficient to permit employees protected by this section to observe a copy on the way to or from their place of work. Premises must be posted not later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license, and for 30 days following license termination.

(2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in appendix D to part 20 of this chapter, by calling (301) 415–5877, via e-mail to *forms@nrc.gov*, or by visiting the NRC's Web site at *http://www.nrc.gov* and selecting forms from the index found on the home page.

(f) No agreement affecting the compensation, terms, conditions, or privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor pursuant to section 211 of the Energy Reorganization Act of 1974, as amended, may contain any provision which would prohibit, restrict, or otherwise discourage an employee from participating in protected activity as defined in paragraph (a)(1) of this section including, but not limited to, providing information to the NRC or to his or her employer on potential violations or other matters within NRC's regulatory responsibilities.

[58 FR 52410, Oct. 8, 1993, as amended at 60 FR 24551, May 9, 1995; 61 FR 6765, Feb. 22, 1996; 68 FR 58809, Oct. 10, 2003; 72 FR 63974, Nov. 14, 2007]

§ 50.8 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act (44 U.S.C. 3501 et seq.). The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number. OMB has approved the information collection requirements contained in this part under control number 3150–0011.

(b) The approved information collection requirements contained in this part appear in §§ 50.30, 50.33, 50.34, 50.34a, 50.35, 50.36, 50.36a, 50.36b, 50.44, 50.46, 50.47, 50.48, 50.49, 50.54, 50.55, 50.55a, 50.59, 50.60, 50.61, 50.62, 50.63, 50.64, 50.65, 50.66, 50.68, 50.69, 50.70, 50.71, 50.72, 50.74, 50.75, 50.80, 50.82, 50.90, 50.91, 50.120, and appendices A, B, E, G, H, I, J, K, M, N,O, Q, R, and S to this part.

(c) This part contains information collection requirements in addition to those approved under the control number specified in paragraph (a) of this section. These information collection requirements and the control numbers under which they are approved are as follows:

(1) In § 50.73, NRC Form 366 is approved under control number 3150–0104.

(2) In § 50.78, Form N–71 is approved under control number 3150–0056.

[49 FR 19627, May 9, 1984, as amended at 58 FR 68731, Dec. 29, 1993; 60 FR 65468, Dec. 19, 1995; 61 FR 65172, Dec. 11, 1996; 62 FR 52187, Oct. 6, 1997; 67 FR 67099, Nov. 4, 2002; 68 FR 19727, Apr. 22, 2003; 69 FR 68046, Nov. 22, 2004; 70 FR 61887, Oct. 27, 2005]

§ 50.9 Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.

(b) Each applicant or licensee shall notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant or licensee violates this paragraph only if the applicant or licensee fails to notify the Commission of information that the applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

[52 FR 49372, Dec. 31, 1987]

REQUIREMENT OF LICENSE, EXCEPTIONS

§50.10 License required; limited work authorization.

(a) *Definitions.* As used in this section, *construction* means the activities in paragraph (a)(1) of this section, and does not mean the activities in paragraph (a)(2) of this section.

(1) Activities constituting construction are the driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of foundations, or in-place assembly, erection, fabrication, or testing, which are for:

(i) Safety-related structures, systems, or components (SSCs) of a facility, as defined in 10 CFR 50.2;

(ii) SSCs relied upon to mitigate accidents or transients or used in plant emergency operating procedures;

(iii) SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function;

(iv) SSCs whose failure could cause a reactor scram or actuation of a safety-related system;

(v) SSCs necessary to comply with 10 CFR part 73;

(vi) SSCs necessary to comply with 10 CFR 50.48 and criterion 3 of 10 CFR part 50, appendix A; and

(vii) Onsite emergency facilities, that is, technical support and operations support centers, necessary to comply with 10 CFR 50.47 and 10 CFR part 50, appendix E.

(2) Construction does not include:

(i) Changes for temporary use of the land for public recreational purposes;

(ii) Site exploration, including necessary borings to determine foundation conditions or other preconstruction monitoring to establish background information related to the suitability of the site, the environmental impacts of construction or operation, or the protection of environmental values;

(iii) Preparation of a site for construction of a facility, including clearing of the site, grading, installation of drainage, erosion and other environmental mitigation measures, and construction of temporary roads and borrow areas;

(iv) Erection of fences and other access control measures;

(v) Excavation;

(vi) Erection of support buildings (such as, construction equipment storage sheds, warehouse and shop facilities, utilities, concrete mixing plants, docking and unloading facilities, and office buildings) for use in connection with the construction of the facility;

(vii) Building of service facilities, such as paved roads, parking lots, railroad spurs, exterior utility and lighting systems, potable water systems, sanitary sewerage treatment facilities, and transmission lines;

(viii) Procurement or fabrication of components or portions of the proposed facility occurring at other than the final, in-place location at the facility;

(ix) Manufacture of a nuclear power reactor under a manufacturing license under subpart F of part 52 of this chapter to be installed at the proposed site and to be part of the proposed facility; or

(x) With respect to production or utilization facilities, other than testing facilities and nuclear power plants, required to be licensed under Section 104.a or Section 104.c of the Act, the erection of buildings which will be used for activities other than operation of a facility and which may also be used to house a facility (*e.g.*, the construction of a college laboratory building with space for installation of a training reactor).

(b) *Requirement for license.* Except as provided in §50.11 of this chapter, no person within the United States shall transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, or use any production or utilization facility except as authorized by a license issued by the Commission.

(c) *Requirement for construction permit, early site permit authorizing limited work authorization activities, combined license, or limited work authorization.* No person may begin the construction of a production or utilization facility on a site on which the facility is to be operated until that person has been issued either a construction permit under this part, RAS-1571

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 16
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

DOCKETED
USNRC
September 9, 2009 (11:00am)
OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

7632-7
POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-55

PART 1 INITIATION

INITIATOR

A. ISSUE, OBSERVATION, OR CONCERN

The video tape of CRDM nozzles inspection (below the RV head Insulation) shows several patches of boric acid accumulation on the RV head. Also one of the CRDM nozzle #67 (P6) shows rust or brown stained boron at the bottom of nozzle where it meets the head. The head area in this vicinity also has rust or brown stained boron accumulation. The video tape of CRDM flanges inspection was reviewed to determine the flange leakage. The inspection of CRDM nozzle #67 flange did not show any leakage during cycle 10 which indicates that the leakage marks and boron accumulation on ~~#67~~ CRDM nozzle #67 are due to leakage from previous operating cycles.

NG-EN-00324 Rev 1 (Boric Acid corrosion Control) outlines several steps to help identify the scope of the problem. Following are some of the steps:

a) The total amount of boron deposits and the amount of boron on each

☐ Q-HOLD TAG APPLIED ☒ CONTINUED

INITIATOR (Print)	SIGNATURE	ORGANIZATION	PHONE NO.
P. K. GOYAL	PKGoyal	DEME	7351

SUPERVISOR

C. PCAQR SUBJECT

BORIC ACID ON RX VESSEL HEAD

D. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
J. HARTIGAN	J Hartigan	7355	4/21/96

PART 2 STATUS AND REPORTABILITY

SHIFT SUPERVISOR

A. MODE / POWER	B. REPORTABILITY	C. OPERABLE
60%	☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A	☐ YES ☐ NO ☒ NON T.S.

D. IMMEDIATE ACTION TAKEN

Condition addressed is programatic vice hardware, therefore part 2C marked NON TS. No immediate actions taken.

☐ CONTINUED

E. SHIFT SUPERVISOR (Signature)	DATE	TIME
Mike Roder	4/21/96	0713

[150]

 NRC001-1629

TEMPLATE = SECY 028 DS 02

ED7632-7

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-551	PAGE 1

PART 1 — INITIATION

A. ISSUE, OBSERVATION, OR CONCERN ☐ PCAQR ADDENDUM

INITIATOR

The video tape of CRDM nozzles inspection (below the RV head Insulation) shows several patches of boric acid accumulation on the RV head. Also one of the CRDM nozzle # 67 (P6) shows rust or brown stained boron at the bottom of nozzle where it meets the head. The head area in this vicinity also has rust or brown stained boron accumulation. The video tape of CRDM flanges inspection was reviewed to determine the flange leakage. The inspection of CRDM nozzle #67 flange did not show any leakage during cycle 10 which indicates that the leakage marks and boron accumulation on ~~#67~~ the CRDM nozzle #67 are due to leakage from previous operating cycles.

NG-EN-00324 Rev 1 (Boric Acid corrosion Control) outlines several steps to help identify the slope of the problem. Following are some of the steps:

a). The total amount of boron deposits and the amount of boron on each

☐ Q-HOLD TAG APPLIED ☒ CONTINUED

B. INITIATOR (Print)	SIGNATURE	ORGANIZATION	PHONE NO.
P. K. GOYAL	PKGoyal	DEME	7351

SUPERVISOR

C. PCAQR SUBJECT

BORIC ACID ON RX VESSEL HEAD

D. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
J. HARTIGAN	J Hartigan	7355	4/21/96

PART 2 — STATUS AND REPORTABILITY

A. MODE / POWER	B. REPORTABILITY	C. OPERABLE
6/0%	☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A	☐ YES ☐ NO ☒ NON T.S.

SHIFT SUPERVISOR

D. IMMEDIATE ACTION TAKEN

Condition addressed is programatic vice hardware, therefore part 2C marked NON T.S. No immediate actions taken.

☐ CONTINUED

E. SHIFT SUPERVISOR (Signature)	DATE	TIME
Mike Boden	4/21/96	0713

[150]

Exhibit 5
Page 1 of 36 Pages

NRC001-1629

000565 GG0151

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT CONTINUATION SHEET

ED 7632B-1

PCAQR NO 96-551 PAGE 2 OF

PART	ITEM	CONTINUATION
1	A	component should be estimated.
		b) The area of the identified boron build-up should be inspected to verify that the boron is localized to the identified area. The area should be inspected to determine if boric acid could have entered the internals of a component and spread internally to a location that is not visible and is susceptible to boric acid corrosion.
		c) The affected area should be inspected to identify any sign of corrosion. If corrosion is present, the amount of corrosion should be estimated. This should include an estimate of corrosion products present as well as an estimate of base metal removed.
		Concern is that above steps may not have been followed to identify the scope of problem.

☐ CONTINUED

ED7632.F-3

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-0551	PAGE 2

PART 3 **REPORTABILITY**

A. EVALUATOR
LCN

DUE DATE 4/26/96

B. REPORTABLE
☒ NO ☐ YES, REPORT NUMBER ____________

C. JUSTIFICATION FOR BEING REPORTABLE
Program administrative issue

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APR 24 1996
PCAQRB

☐ CONTINUED

D. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
E. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
P. W. Smith	[signature]	7744	4/23/96

REGULATORY AFFAIRS

[152]

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-551

PAGE 3

PART 4 INITIAL ASSESSMENT

A. EVALUATOR: D E M E

DUE DATE: 4/29/96

B. CLASSIFICATION / APPLICABILITY

☒ Q AQ ☐ NQ ☒ ASME ☐ NON-ASME

PLANT SUBSYSTEM 0 6 2 0 1 EQUIPMENT ID RV Head

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C. SIGNIFICANCE

Process As Other Work / Action Document

☐ Category 4 Document Initiated

Process As PCAQR

☐ Category 3 Weighting Factor ~~76.5~~ 202 5-6-96 CATPR ☐ NO ☒ YES

☒ Category 2

☐ Category 1 QC ↓

D. DUE DATE

Part 5 5 / 10 / 96 Part 6 6 / 3 / 96 kna 5-6-96

E. JUSTIFICATION

A walkdown inspection of RV head is performed during each outage in response to NRC Generic Letter 88-05 (Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants). The walkdown inspection includes the visual inspection of CRDM flange area. In addition RV head is inspected for boric acid deposits. The safety evaluation submitted to NRC for B&W CRDM nozzle cracking issue takes credit of this inspection. The basis being if there is a CRDM nozzle crack, the primary coolant escaping from through-wall crack will exit from RV head pentration in the form of flashing borated steam and/or boric acid crystals (snow) which will continue to deposit on RV head throughout the operating cycle. This deposit can be detected during the head inspection at the end of cycle and corrective action(s) taken. Since the boric acid

☐ Material Deficiency Tag Installed ☒ CONTINUED

F. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
P.K. GOYAL	PKGoyal	7351	4/29/96
G. SUPERVISOR (Print)	**SIGNATURE**	**PHONE NO.**	**DATE**
J. P. HARTIGAN	J Hartigan	7355	4/29/96

H. MANAGEMENT CONCURRENCE - CATEGORY 4, 2 and 1 (Signature)

Robert E Donnelly 4/29/96 (COMMENTS ON CONTINUATION SHEET)

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature) DATE 5/7/96

[153]

NRC001-1632

POTENTIAL CONDITION
ADVERSE TO QUALITY REPORT CONTINUATION SHEET
ED 7632E-1

PCAQR NO 96-551

PAGE 3a OF

PART	ITEM	CONTINUATION
4	E	deposits are not cleaned it is difficult to distinguish whether the deposits occurred because of the leaking flanges or the leaking CRDM. This situation represents an adverse trend with the potential for greater than marginal consequences (Cat 2, C.3 Attachment 2 of NG-NA-0702) The peripheral nozzles on the downhill side of the RV head have the high potential for cracking. The nozzle #67 mentioned in this PCAQR is a downhill side nozzle. The CRDM nozzles are attached to the RV head by an interference fit and there is no weld at the top of head. Thus coolant from leaking flanges will travel down to head and can enter in the head via the upper counterbore area (See Fig 1) and initiate internal corrosion. The visual inspection can not determine whether the coolant has gone inside the heads or not. Also for the step 6 of NG-EN-00324 "The area should be inspected to determine if boric acid could have entered the internals of a component and spread internally to a location that is not visible" can not be completed. This represent a situation that could have escaped detection by visual examination. (Cat 2, C.5 Attachment 2 of PCAQR procedure NG-NA-00702.
		[signature] 5/3/96
		I have signed the part 4E justification and concurrence as a conservative measure. Nozzle cracking is, of course, a significant issue. However, at present, the probability of occurrence is relatively low. We should remove boron from the reactor pressure vessel head as best we can and so as to minimize dose. This will enable us to monitor any leakage, should a nozzle crack initiate. I believe that it is questionable that boron would enter the nozzle area from the outside because of the head temperature and the fact that there is an interference fit. I also do not believe that the vessel head area is non-conforming.
		Robert E Donnellon 4/29/96

☐ CONTINUED

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POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-551	PAGE 4

PART 5 — REMEDIAL ACTION

A. EVALUATOR	DUE DATE
DEME	5/10/96

EVALUATOR

B. QUALITY CHARACTERISTICS AFFECTED

1) The boron deposits around CRDM nozzle #67 and over the RV head can cause corrosion of Carbon Steel RV head. This corrosion if left unchecked could impact the RCS pressure boundary.

2) There is an industry wide concern regarding CRDM nozzle cracking due to PWSCC. This concern is addressed by RV head inspection. The inspection is done to check the CRDM nozzle cracking which can be ☒ CONTINUED

C. APPARENT CAUSE

1. Leaking CRDM flanges
2. Lack of removal of boron deposits from the RV head after each operating cycle.

☐ CONTINUED

D. SYSTEM CAPABLE OF PERFORMING SPECIFIED FUNCTION

☐ N / A - Non Tech Spec and not a potentially reportable system

☒ YES - Provide Justification unless Weighting Factor less than 11

☐ NO - Initiate an addendum if Part 2 is Operable / Yes

JUSTIFICATION

The two concerns identified in this PCAQR are the boron deposits around nozzle #67 and on the RV head. The boron has been removed around nozzle #67 and there was no visible appearance of corrosion at this location. (Nozzle itself is not impacted because it is constructed of Alloy 600 & Alloy 600 is not impacted by boric acid) ☒ CONTINUED

E. EXTENT OF CONDITION ☐ N / A If Weighting Factor less than 16

The extent of condition is limited to RV head because all of the boron deposits can not be removed due to limited accessibility to head area. Normally the boron deposits from the components are removed and surfaces cleaned.

☐ CONTINUED

F. 10CFR Part 21 REPORTABLE ☒ NO ☐ YES - INITIATE AN ADDENDUM

[initials] 5-13-96

per Prasoon Goyal.

[155]

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT CONTINUATION SHEET

ED 7632E-1

PCAQR NO 96 – 551 | PAGE 5 OF

PART	ITEM	CONTINUATION
5	B	detected by boron deposits on the head. The existing boron deposits make it very difficult to draw any conclusion from the inspection.
5	D	There could also be corrosion damage within the reactor vessel head penetration due to boric acid corrosion resulting from a through wall crack in the CRDM nozzle. The possibility of this type of crack is extremely low since no large accumulation of boron was found on the head. (A leak rate of .0417 gpm of 600 ppm boron solution will correspond to an accumulation of 627 lb/year of boric acid crystal). Thus RCS pressure boundary is not impacted.
		There are some boron deposits on the head. These boron deposits will create negligible corrosion rate since the head temperature is ≥ 550°F. This is based on the result of B&W boric acid corrosion test program performed for B&WOG (B&W Document # 51-1229638-1). The testing showed that the highest corrosion rate for carbon steel occurred at approximately 300°F, at the interface where the dried boric acid crystals were re-wetted by the leakage. Lower levels of corrosion occurred at temperatures approaching 500°F. Almost no corrosion occurred at temperature greater than 550°F. The only time the higher corrosion rate will be encountered is during shutdown and heatup when the temperature of head will be well below 550°F. Since this duration is very short no impact on RV head is anticipated. Therefore system will continue to perform its intended function.

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☐ CONTINUED

[illegible]

ED7632A (Continued)

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-0551	PAGE 6

PART 8 — REMEDIAL ACTION (Continued)

G. SPECIFIC ACTION NECESSARY TO CORRECT CONDITION (Restoration)

1. Remove boron deposits from CRDM nozzle #67
2. Remove boron deposits from the RV head.

Record the above on a video tape for future use/basis.

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☐ CONTINUED

EVALUATOR

H. JUSTIFICATION FOR CORRECTIVE ACTIONS

The removal of boron deposits from nozzle #67 and the RV head would provide a basis for comparison of boron buildup after the completion of next operating cycle.

☐ CONTINUED

I. CORRECTIVE ACTION ORGANIZATION	DUE DATE
RP - remove boron from nozzle #67 and RV head [Note: The boron from RV head was removed to the extent practical considering cleaning equipment limitations, size of mouse holes and dose]	5/1/96 A

☐ CONTINUED

J. ASME CODE	DATE
☐ NO ☒ YES, ANI Review [signature]	5-9-96

K. DISPOSITION
☐ N/A ☐ REJECT ☒ REWORK ☐ USE-AS-IS ☐ REPAIR

L. ENGINEERING APPROVAL (ASME, USE-AS-IS, REPAIR)	DATE
N/A	

M. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
P.K. GOYAL	[signature]	7351	5/9/96
N. SUPERVISOR (Print)	**SIGNATURE**	**PHONE NO.**	**DATE**
J. HARTIGAN	J Hartigan	7355	5/10/96
O. PCAQR REVIEW BOARD		**DATE**	**CATPR**
Robert B Cond		11/18/98	☐ NO ☒ YES

[157]

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-0551

PAGE 7

EXPERIENCE REVIEW

A. EVALUATOR: QC

CATPR RECOMMENDED: ☐ YES ☒ NO JDL 5-6-96

DUE DATE

B. SIMILAR EVENTS

The INPO keyword index was searched for documents that discuss related events. The following documents may be applicable:

- SER 20-93, "Intergranular Stress Corrosion Cracking in Control Rod Drive Mechanism Penetrations," and
- IN 90-10, "Primary Water Stress Corrosion Cracking of Inconel 600."

The PCAQR database was searched for previous problems with reactor vessel head inspections under procedure NG-EN-00324. No previous problems were found.

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☐ CONTINUED

EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
Jerry D. Lammlee	Jerry D. Lammlee	7362	5-6-96

ASSIGNED ORGANIZATION

[158]

NRC001-1637

E97632-N

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-551	PAGE 8

WORK COMPLETION

A. ☒ REMEDIAL CORRECTIVE ACTION ☐ CATPR

B. ASSIGNED ORGANIZATION: DEMS

DUE DATE: 9/30/96

ASSIGNED ORGANIZATION

C. ASSIGNED ACTION

1) Please Document (or reference the quality document) the estimate of the total amount of boron deposits & amount of boron on each component.

2) Please provide an evaluation of the a) area of the boron buildup to ensure that the boron is localized to the identified areas & b) determination of if boric acid could have entered the internals of a component and spread internally to a location that isn't visible and is susceptable to boric acid corrosion.

☐ CONTINUED

D. RESPONSE

See attached.

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☐ CONTINUED

E. COMPLETED BY (Print)	SIGNATURE	PHONE NO.	DATE
P. K. GOYAL	PKGoyal	7351	11/22/96
F. SUPERVISOR (Print)	**SIGNATURE**	**PHONE NO.**	**DATE**
T S SWIM	T S Swim	7299	11/22/96

[159]

POTENTIAL CONDITION
ADVERSE TO QUALITY REPORT CONTINUATION SHEET
SD 1605E.1

PCAQR NO 96-551

PAGE 9 OF

PART	ITEM	CONTINUATION
C	1	This PCAQR is the quality document which recorded the boric acid deposit on the RV head. The deposits were discovered during the visual inspection of the RV head performed through the mouseholes utilizing a video camera. The extent of the inspection was limited to approximately 50 to 60% of the head area because of the restrictions imposed by the location and size of mouseholes. The inspection showed varying sizes of boric acid mounds scattered in various areas of head. It is extremely difficult to ~~data~~ develop an estimate of the amount of boric acid deposit because of the deposit scatter and limited inspection. It should be noted that during 8th refueling outage the boric acid deposits were removed (to the extent possible) by washing the RV head. Also boron deposits which were accessible, were removed last outage (10RFO). Other B&W plants (except DB & ANO1) have large access holes (≈10"φ) in the service structure for inspection and removal of boron deposits. If one could remove all the boron deposits from RV head, the issue of ongoing corrosion will go away.
	2. a)	The boron deposits are limited to the RV head area in the ~~visit~~ vicinity of various CRDMs.
	b)	The area which could be cleaned did not show any significant corrosion. The condition of the area from which boron could not be removed is not known. But is anticipated that the corrosion of this area should be minimal since the head temperature is greater than 550°F.

☒ CONTINUED

[160]

NRC001-1639

POTENTIAL CONDITION
ADVERSE TO QUALITY REPORT CONTINUATION SHEET
SD 75328-1

PCAQR NO 96-551 | PAGE 10 OF

PART	ITEM	CONTINUATION
c	b	Contd:

The boric acid corrosion rate on carbon steel is almost negligible at temperature greater than 550°F. The highest corrosion rate occurs around 300°F with continuous rewetting of the surface by leakage. The issue is whether boric acid could have entered the internals of a component and spread internally to a location that is not visible and susceptible to corrosion. The boric acid could enter the inside of RV head only through openings for CRDMs. Since the top counterbore (see Fig-1 attached to this PCAQR) has a radial clearance of approximately .007 inch during the plant operation (Ref SE for B&W Design RV head CRDM nozzle cracking BAW-10190P), it is possible for boric acid to seep through this clearance. But possibility of this occurring is quite low because of the very small gap and high temperature. This could happen if the conditions are just right e.g already existing boric acid deposits around CRDMs, low temperature and leaking CRDM flanges. This type of leakage damage is extremely difficult to measure because area of interest can not easily be inspected.

Note: This PCAQR was written because steps outlined in NG-EN-00324 Rev 1 (Boric Acid Corrosion Control) can not be fully implemented.

[signature] 11/22/96.

[signature] 11/22/96

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☐ CONTINUED

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT CONTINUATION SHEET
ED 7032G

PCAQR NO. 96-551 | PAGE 11 OF

PART	ITEM	CONTINUATION
6	B	
	TO:	QC
	DUE:	July 26, 1996
	ACTION	Complete the Root Cause/CATPR according to the action plan.
		PCAQRB Chairman John Johnson June 11, 1996

☐ CONTINUED

[162]

NRC001-1641

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-0551

PAGE 12

PART 6 CAUSE / CORRECTIVE ACTION TO PREVENT RECURRENCE

A. ☐ APPARENT CAUSE ☒ ROOT CAUSE ☐ MULTI-DISCIPLINE ROOT CAUSE

B. EVALUATOR QC

DUE DATE 1/31/97

C. CAUSES / CONTRIBUTING FACTORS

the cause of the inability to carry out the actions in DM 96-EN-00324 is due to inaccessability of areas for inspection. In the part 4 response, as well as in the Part 5, the limited inspection area poses a condition that cannot be evaluated. Mechanical connections can + do leak, & inspection of all suseptable areas are needed.

Acknowledged PCAQRB John Johnson 5/6/97

☐ CONTINUED

EVALUATOR

D. CORRECTIVE ACTION TO PREVENT RECURRENCE

Enlarge the inspection holes to permit inspection of all suseptable areas + to inspect the CRDM nozzle penetrations of the head. Additional holes will be needed for access to the upper part of head

☐ CONTINUED

E. JUSTIFICATION FOR CORRECTIVE ACTION

this will ensure that inspection for boron deposits + subsequent removal

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☐ CONTINUED

F. CORRECTIVE ACTION ORGANIZATION

DUE DATE

Corrective action concurrence needed by the PCAQRB - Please schedule board meeting for discussion.

☐ CONTINUED

G. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
H. SUPERVISOR (Print) D. SCHREINER	SIGNATURE	PHONE NO. 7845	DATE 2-7-97

I. MANAGEMENT CONCURRENCE - CATEGORY 2 & 1 (SIGNATURE) (A)

J. PCAQR REVIEW BOARD

DATE 11/18/98

[163]

NRC001-1642

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT CONTINUATION SHEET

ED 78322-1

PCAQR NO 96-0551 | PAGE 1 2OF

PART	ITEM	CONTINUATION
		Following is the response to the action items assigned to DEMS:
	1)	The system engineer TOM BASIO at Three Mile Island Unit 1 (TMI) was contacted regarding the extent of Reactor Vessel (RV) head cleaning through the large openings in the service structure (SS). According to him TMI is able to reach the entire head including the top of head and clean it. They use long handle scraper to loosen the boric acid deposits and then vacuum the deposits. The cleaning is performed when the head is on the head stand. Note: TMI head inspection is done under the augmented ISI program. Red Emery of Duke Power indicated that they are able to clean the entire RV head through the openings in the service structure. Duke utilizes high pressure water to clean the head. The head is cleaned when it is still on the RV. The system engineer Jim Lene at Crystal River 3 was contacted twice. But he has not responded because the CR is in outage. Based on the above the large openings in SS do provide access for cleaning the entire head.
	3)	The action was to visit TMI 1, if possible, during the Fall 97 refueling outage and evaluate the head inspection capability. I was not able to visit TMI because of the conflicting schedule. However we were able to get a copy of the video tape of the RV head after the cleaning. The tape quality is not very good. But it shows that the RV head does not have any boric acid deposits. The DB system engineer Dan Haley had also viewed this tape.
		[signature] 11/19/97
		[signature] 11/19/97

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☐ CONTINUED

[164]

POTENTIAL CONDITION
ADVERSE TO QUALITY REPORT CONTINUATION SHEET

PCAQR NO.	PAGE
96-551	14 OF

PART	ITEM	CONTINUATION
6	B	

To: SYME

Due: November 21, 1997

Action: 1)Please complete the Root Cause and propose and justify CATPR IAW NG-NA-702.

2)Please obtain Manager concurrence.

PCAQRB John Johnson 5/6/97

☐ CONTINUED

[165]

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 1998-0551

PAGE 15

PART 5 — CAUSE / CORRECTIVE ACTION TO PREVENT RECURRENCE

A. ☐ APPARENT CAUSE ☒ ROOT CAUSE ☐ MULTI-DISCIPLINE ROOT CAUSE

B. EVALUATOR: S Y M E

DUE DATE: 10/15/98

C. CAUSES / CONTRIBUTING FACTORS

The restricted access to the area on the top of the reactor vessel head, below the CRDM flange insulation, has resulted in an inadequate ability to completely inspect and clean the outside/top surface of the reactor vessel head. This surface requires inspection/cleaning due to the potential for boric acid to accumulate on the head, as a result of system leakage and the corrosion concerns that are associated with boric acid leakage.

☐ CONTINUED

D. CORRECTIVE ACTION TO PREVENT RECURRENCE

Modification 94-0025 has been initiated to install 9 inspection/access holes, with removable covers, in the service structure. The access holes will allow both direct and remote visual inspection capabilities. The modification will also allow for adequate access to the top surface of the head to clean/remove any accumulated boric acid buildup. The modification has been approved for implementation during 13 RFO by both the PRC and the WSC

☐ CONTINUED

E. JUSTIFICATION FOR CORRECTIVE ACTION

Installation of the additional access holes in the service structure will provide the access that is needed to inspect and clean the top surface of the reactor head. Inspection/cleaning activities following modification installation will provide baseline information for future inspection purposes. Previous inspections of the vessel head and analysis of the corrosion conditions present has determined that installation of this modification can be scheduled for 13 RFO with very limited risk of damage to the surface of the head from boric acid corrosion.

☐ CONTINUED

F. WORK COMPLETION DOCUMENTS

Modification 94-0025 CATS Item 1

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PCAQRB

Acknowledged SRB Robert B Coad 10/28/98

☐ CONTINUED

G. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
Daniel E. Haley	Daniel E. Haley	7841	09/21/98 10-14-98

H. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
Glenn R. McIntyre	[signature]	8165	09/21/98

I. MANAGEMENT CONCURRENCE

[signature] 9-21-98 R E D[signature] 9/22/98

J. PCAQR REVIEW BOARD

Robert B Coad

DATE 11/18/98

PART	ITEM	CONTINUATION
	B	Action Due November 21, 1997

8/28/97

On Thursday, August 7, 1997, the below listed persons met to discuss this PCAQR. The following actions were determined to be necessary to resolve the issue. These actions are to determine if the Mod will provide the ability to clean the head which is essential to an effective evaluation of any boric acid found on the head. ~~The boron acid presently on the head is believed to have come from leaking CRDMs in past outages, prior to gasket replacements which have eliminated most leakage. But because the head has not been completly cleaned, it is not possible to make a clear determination that we do not have active leakage.~~

Attendees: Prasoon Goyal
John Hartigan
Mike Shepherd
Ed Chimahusky
Dan Haley
Glenn McIntyre

1. Determine how well the head can be cleaned after the installation of the larger access openings. Three other units have installed the proposed Modification and will be contacted to determine if the Mod enables the complete cleaning of the head.
Action: Prasoon Goyal Due: November 1, 1997.

2. Evaluate the possibilities for an alternative head cleaning process. If a suitable process of cleaning the head can be implemented, the Mod would be unnecessary.
Action: Dan Haley, System Engineer Due: November 1, 1997

3. Visit TMI, if possible, during their Fall 97 refueling outage and evaluate the head inspection capability after the installation of larger access ports.
Action: Prasoon Goyal, DEMS Due: November 1, 1997

4. Determine if the Modification is required and present to the PRG or if procedure revisions are required to clarify expectations for the Engineering Evaluations required by NG-EN-00324, Boric Acid Corrosion.
Action: Dan Haley, System Engineer Due: December 1, 1997

The next action is contingent on the above actions.

5. **Either** Present Mod to PRG
Action: Mechanical Systems Engineering Due: February 28, 1998

Or Implement Enhanced Cleaning Process
Action: Mechanical Systems Due: February 28, 1998

Or Revise NG-EN-00324
Action: Mechanical Systems Engineering Due: April 1, 1998

Glenn McIntyre
Supervisor, Mechanical Systems
x8165

PART	ITEM	CONTINUATION
		Continuation of 8/12/97 extension request The following are SYME's responses to the assigned items on the referenced extension request action plan: **2. Evaluation of an alternative head cleaning process.** Due to the limited access to the surface of the reactor vessel head, both visually and physically and to the constraints placed on the inspection/cleaning activities by schedule and ALARA, no new feasible methods of inspection and or cleaning have been found. Discussions with other PWR sites has revealed that most have already modified/enlarged the access openings through the service structure to the upper head surface or are planning a modification to enlarge the access openings in the service structure. **4. Determine if a modification is required or revise procedures to clarify current processes.** Without access to the entire surface of the reactor vessel head it is not possible to perform a complete inspection and cleaning of the head. The limited inspection capabilities that exist have shown that there are some existing boron crystal deposits on the head surface. These deposits cannot be reached for cleaning/removal through the current access holes. It cannot be determined, without some element of doubt, if these deposits are all due to some past leakage source that has been corrected or if there is a new leak present. Modifying the support structure access holes would allow for a complete inspection of the head surface, access to clean the surface of the head to remove any existing boron crystal deposits and should allow for improved future inspection/cleaning schedule times. SYME has determined that the modification to enlarge the access holes in the head support structure should be pursued. Daniel E. Haley 12-17-97 Daniel E. Haley 12-17-97 Glenn R. McIntyre 8165 ☐ CONTINUED

POTENTIAL CONDITION
ADVERSE TO QUALITY REPORT CONTINUATION SHEET

PCAQR NO.	PAGE
96-551	18 OF

PART	ITEM	CONTINUATION
6	To:	SYME
	Due:	December 11, 1998
	Action:	1. The Part 6 response was rejected by the SRB/PCAQRB because it didn't include the required attributes of a full Root Cause Evaluation (e.g., problem statement, event narrative, why previous corrective actions/self assessments were ineffective to prevent recurrence, etc. – see NG-NA-702 step 6.7.2.b). 2. Please complete a Root Cause Evaluation IAW NG-NA-702. 3. Please address the Davis-Besse response to GL 88-05 for the evaluation of the significance of boric acid on the head. 4. Please obtain manager concurrence. SRB/PCAQRB Rob B Cie October 28, 1998

☐ CONTINUED

[169]

ED 7630H-3 (W)

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO 1998-0551	PAGE ~~1~~ 19

KW 11-9-98

PART 4 **INITIAL ASSESSMENT**

A. ORGANIZATION: S Y M E — DUE DATE 10/??/98

B. CLASSIFICATION / APPLICABILITY
☒ ASME ☐ NON-ASME

C. SIGNIFICANCE

Process As Other Work / Action Document

☐ Category 4 Document Initiated ______

RECEIVED
OCT 2 9 1998
PCAQRB

Change to MRC Recommendation

☐ Category 3
☒ Category 2 CATPR ☐ NO ☒ YES
☐ Category 1

D. DUE DATE
Part 5 / / Part 6 10/15/98

E. COMMENTS

This PCAQR Part 4 is being submitted to propose an Apparent Cause Root Cause instead of a Root Cause. The condition identified by this PCAQR was that an engineer could potentially not adequately view portions of the reactor vessel head to evaluate for Boric Acid Corrosion concerns. The depth of investigation to determine why he could not see does not warrant a ROOT CAUSE. He couldn't view all of the vessel head area because the access holes to do so are not big enough to allow it. The attributes to be gained by completion of a full blown Root Cause will not change the corrective actions, which are to make bigger holes or more adequately utilize the existing holes. An event narative, extent of condition, and previous occurrence evaluation will not add any observable quality to this situation. This issue addresses the software issue of inspection, not the hardware issue of head leakage. An apparent cause will more than adequately support Corrective Action to Prevent Occurrence.

Approved: [signature] 10-28-98
J. W. Rogers Manager — Date

F. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
Glenn McIntyre	[signature]	8165	10-29-98
G. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
Glenn McIntyre	[signature]	8165	10-29-98
H. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	John Johnson [signature]		DATE 11/?/98

EVALUATOR

NUCLEAR SAFETY AND INSPECTION

QUALITY REPORT (PCAQR)	1996-0551	

PART 7 VERIFICATION

	YES	NO	N/A
A. APPROVED REMEDIAL ACTION HAS BEEN INITIATED AS DESCRIBED	X		
B. APPROVED CORRECTIVE ACTION TO PREVENT RECURRENCE HAS BEEN IMPLEMENTED AS DESCRIBED	X		
C. Q-HOLD TAGS HAVE BEEN REMOVED			X
D. CONDITIONAL RELEASE HAS BEEN RESOLVED			X
E. STOP WORK ACTION HAS BEEN WITHDRAWN			X

F. DOCUMENTS REVIEWED

CATS

☐ CONTINUED

G. COMMENTS

NA

☐ CONTINUED

H. QA REVIEW OF AUDIT / SURVEILLANCE PCAQR			DATE
NA			
I. REGULATORY AFFAIRS REVIEW OF REPORTABLE PCAQR			DATE
NA			
J. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
Earl E. Murphy	Earl E. Murphy	8510	01-19-99
K. NSI MANAGER CONCURRENCE OF UNACCEPTABLE PCAQR			DATE
NA			

70321-3

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-0551

PAGE

EXTENSION REQUEST

A. ASSIGNMENT
Root Cause / CATPAK

☐ CONTINUED

B. ORGANIZATION
SYME

DUE DATE
5-1-97

C. PCAQR CATEGORY
2

EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4)
11-21-97

D. REASON FOR EXTENSION REQUEST
SYME has agreed to accept this PCAQR from QC for resolution. I have reviewed the PCAQ and outstanding issues. Resolution will either result in a Modification or procedural guidance changes which will outline acceptable inspections.

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)
The concerns identified are not a concern in the short term. The procedure is only used for Reactor vessel head inspections during the outage. Completion by 11-21-97 will allow procedure changes to be implemented prior to 11RFO. The Modification if approved does not require implementation in 11RFO because the potential damage is not a short term problem.

☐ CONTINUED

F. REQUESTOR (Print) GLENN R. McINTYRE	SIGNATURE	PHONE NO.	DATE
G. SUPERVISOR (Print) GLENN R. McINTYRE	SIGNATURE	PHONE NO. 8165	DATE 4-28-97

H. REMARKS
☐ PREVIOUS EXTENSIONS
N/A
Acknowledged by PCAQRB 5/6/97

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)
Not used.

DATE

REQUE

[172]

NRC001-1651

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-0551	PAGE

EXTENSION REQUEST

REQUESTOR

A. ASSIGNMENT

Complete Root Cause and CATPR and get Manager concurrence

☐ CONTINUED

B. ORGANIZATION	DUE DATE
S Y M E	11/21/97

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL
2	December 17, 1997

D. REASON FOR EXTENSION REQUEST

Provide sufficient time to complete the attached action plan which will lead to a determination of whether or not a modification should be pursued or the procedure revised or potentially an option to perform an enhanced cleaning without the Mod.

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

The next inspection will not be performed until 12RFO. This delay will provide additional information, to make a more informed decision without impacting implementation. Based on the input from the involved engineers this is acceptable and is a delay of only 1 month. There will be no impact on hardware, programs or activities.

☐ CONTINUED

	SIGNATURE	PHONE NO.	DATE
F. REQUESTOR (Print) Glenn R. McIntyre	[signature]	8165	8-12-97
G. SUPERVISOR (Print) Joseph W. Rogers	[signature]	7302	8-12-97

H. REMARKS

1 ~~NO~~ PREVIOUS EXTENSIONS

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	DATE
[signature]	8/12/97

[181]

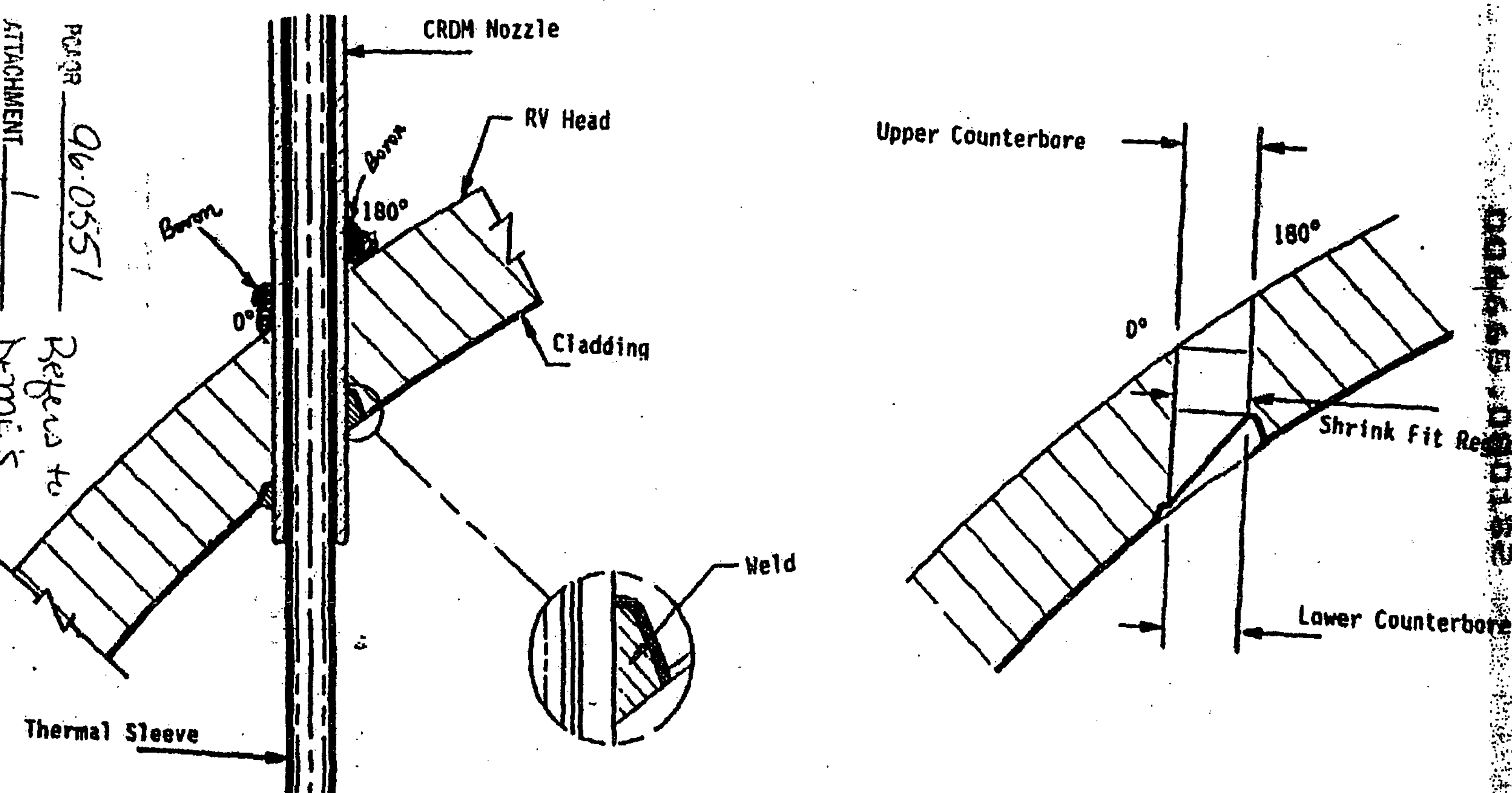
NRC001-1652

PCAQR 96-0551
ATTACHMENT 1
PAGE 1 OF 1

Refers to Demi 5 4/24/96 Response

FIGURE [illegible]-1

B&W-DESIGN CRDM NOZZLE



(A) View of Nozzle with Thermal Sleeve

(B) View of Penetration and Counterbores

ACTION PLAN

38 4488

PLAN NUMBER	PAGE of
DATE PREPARED	PREPARED BY

TITLE Root Cause /CATPR for PCAOR 96-551

SPECIFIC OBJECTIVE

STEP NUMBER	ACTION STEPS	PRIME RESPONSIBILITY	ASSIGNED TO	START DATE	TARGET DATE	DATE COMPLETED
1	MEET WITH DESIGN TO DISCUSS + CONTRAST PRIOR ATTEMPTS TO SPONSOR MOD. THRU WORK SCOPE/ PRG	[illegible signature]			6-24-96	
2	REVIEW PAST MWO'S + WORK REPORTS (from B+W) that will aid in determining adverse conditions.	[illegible signature]			7-8-96	
3	Develop corrective action based on above.	[illegible signature]			7-22-96	

Return to QC's 6/10/96 information request

PCAOR 96-0551
ATTACHMENT 2
PAGE 1 OF 1

[183]

NRC001-1654

RECORD END SHEET

END OF RECORD

[184]

	PCAQR		4/21/1996		Closed
	1996-0551		1/19/1999		PK GOYAL
	No		4/21/1996		DEMS
			7:13:00 AM		
	2				

SEVERAL PATCHES OF BORIC ACID ACCUMULATION WERE FOUND ON THE REACTOR HEAD.

1/20/99
Downgraded to CAT. 2 - Still send to SRB.

	IMPLEMENT MODIFICATION 94-0025 DURING 13 RFO	C	c	S

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO.	PAGE
96-0551	

EXTENSION REQUEST

Complete root cause, CATPR and obtain Manager concurrence.

☐ CONTINUED

B. ORGANIZATION	DUE DATE
S Y M E	~~11/21/97~~ 12/17/97

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL
2	March 20, 1998

D. REASON FOR EXTENSION REQUEST

The 8/12/97 extension request action plan has reached a decision point to proceed with a proposal to modify the reactor head support structure. This extension will allow time for the modification process to be started and a presentation to the PRG scheduled. PRG approval of the modification will substantiate the modification as a CATPR for this PCAQR.

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

This is a non reportable condition and not a non conformance with respect to the reactor head. The boric acid corrosion process is a slow acting process and does not pose any immediate safety concern for the near short term.

☐ CONTINUED

REQUESTER

F. REQUESTER (Print)	SIGNATURE	PHONE NO.	DATE
Daniel E. Haley	Daniel E. Haley	7841	12/17/97
G. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
Glenn R. McIntyre	[signature]	8165	12/17/97

PCAQRB

H. REMARKS

EXTENSIONS 3

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	DATE
John Johnson	1/12/98

[173]

NRC001-1657

ED 7##20-8 (##)

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 98-551		PAGE

EXTENSION REQUEST

A. Assignment

Complete root cause, CATPR and obtain Manager concurrence *and director. Kmr 3-10-98*

B. ORGANIZATION	DUE DATE
S Y M E	03/20/98

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL
~~2~~ 1 *Kmr 3-10-98*	October 15,1998

D. REASON FOR EXTENSION REQUEST

A decision point has been reached to proceed with a proposal to modify the reactor head service structure. A presentation of the proposed modification was included on the agenda for the PRG meeting that was scheduled for 3/13/98. The PRG meeting on 3/03/98 was cancelled and the proposal could not be presented. Due to RFO 11 schedule constraints, the PRG will not meet again until after the outage. The 3/20/98 due date for this PCARQ can not be met, the extension request will allow time to schedule the presentation after the refueling outage.

E. JUSTIFICATION FOR DELAY (Address Effect On Hardware, Programs, and Activities)

This is a non reportable condition and not a nonconformance with respect to the reactor head. The boric acid corrosion process is a slow acting process and does not pose any immediate safety concern for the near short term.

F. REQUESTER (Print)	SIGNATURE	PHONE NO.	DATE
Daniel E. Haley	*Daniel E. Haley*	7841	03/09/98
G. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
Glenn R. McIntyre	*[signature]*	8165	03/09/98

H. REMARKS

EXTENSIONS *3*

R E Q U E S T E R

P C A Q R B

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)

J. W. Stevens, Jr *3/18/98*

[174]

NRC001-1658

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-702 / 551

PAGE

EXTENSION REQUEST

A. ASSIGNMENT: Complete Root Cause

☐ CONTINUED

B. ORGANIZATION: QC

DUE DATE: 1-1-97

C. PCAQR CATEGORY: 2

EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4): 1-31-97

D. REASON FOR EXTENSION REQUEST

Availability of personnel over the holidays

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

Not reportable - no non-conforming condition

☐ CONTINUED

F. REQUESTOR (Print)	SIGNATURE	PHONE NO.	DATE

G. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
Dennis Schelling	[signature]	7845	12/30/96

H. REMARKS

2 ☐ PREVIOUS EXTENSIONS 3

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature): G. Johnson / kmm

DATE: 12-30-96

REQUE

[175]

NRC001-1659

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-551	PAGE

EXTENSION REQUEST

A. ASSIGNMENT Complete Root Cause

☐ CONTINUED

B. ORGANIZATION	DUE DATE
Q C	12-1-96

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4)
2	1-1-97

D. REASON FOR EXTENSION REQUEST

I've been too busy with the ISRO investigation to get w/ Design to discuss limitations/CA for RV head leaks

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

Not reportable, no non-conforming condition.

☐ CONTINUED

REQUESTOR

F. REQUESTOR (Print)	SIGNATURE	PHONE NO.	DATE
G. SUPERVISOR (Print)	**SIGNATURE**	**PHONE NO.**	**DATE**
Dennis Schreiner		7845	11-27-96

H. REMARKS

4 ☐ PREVIOUS EXTENSIONS

☐ CONTINUED

I. CONCURRENCE (PCAQR REVIEW BOARD CHAIRMAN (Signature))	DATE 4/29/97

[176]

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO.	PAGE
96-551	

EXTENSION REQUEST

A. ASSIGNMENT

Complete Root cause

☐ CONTINUED

B. ORGANIZATION	DUE DATE
QC	9-30-96

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4)
2	12-1-96

D. REASON FOR EXTENSION REQUEST

Third ~~[illegible]~~ overdue notice sent - even though Cat. 2 - this is low priority work given no non-conforming hardware issue exist

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

This condition is not reportable + engineering has stated there is no non-conforming condition on the RV head

☐ CONTINUED

	SIGNATURE	PHONE NO.	DATE
F. REQUESTOR (Print)			
G. SUPERVISOR (Print) Dennis Schreiner	[signature]	7845	10-16-96

H. REMARKS

3 ☐ PREVIOUS EXTENSIONS This extension approved based on the 8/20/96 PCAQRB direction that it doesn't matter how long this will take.

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	DATE
[signature]	10/17/96

REQUE

[177]

NRC001-1661

NG-7NA-1

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 96-551	PAGE

EXTENSION REQUEST

A. ASSIGNMENT

1. Document or reference the quality document for estimate of total amount of boron deposit
2. Evaluation of the area of boron buildup

☐ CONTINUED

B. ORGANIZATION	DUE DATE
D E M S	9/30/96

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4)
2	11/22/96

D. REASON FOR EXTENSION REQUEST

Higher priority work (USAR review) and additional time is required to research the disposition of boron deposits on RV head during the previous outages.

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

See attached 5D response.

Note: Since the boric acid corrosion process is a slow process, there is no immediate safety concern. However over a long period of time it can cause damage if no corrective action(s) is taken.

☐ CONTINUED

REQUESTOR

F. REQUESTOR (Print)	SIGNATURE	PHONE NO.	DATE
P.K. GOYAL	PKGoyal	7351	9/30/96
G. SUPERVISOR (Print)	**SIGNATURE**	**PHONE NO.**	**DATE**
T S SWIM	T S Swim	7799	9/30/96

H. REMARKS

1 ☐ PREVIOUS EXTENSIONS

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	DATE
John Johnson	10/4/96

[178]

NRC001-1662

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-551

PAGE

EXTENSION REQUEST

A. ASSIGNMENT

Complete Root Cause

☐ CONTINUED

B. ORGANIZATION

QC

DUE DATE 7/26/96

C. PCAQR CATEGORY 2

EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4) 9-30-96

D. REASON FOR EXTENSION REQUEST

Work load has precluded spending the necessary time to gather info needed

☐ CONTINUED

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

THIS CONDITION IS NOT REPORTABLE + ENGINEERING HAS STATED THERE IS NO NON-CONFORMANCE WITH RESPECT TO THE RV HEAD

☐ CONTINUED

F. REQUESTOR (Print)	SIGNATURE	PHONE NO.	DATE
G. SUPERVISOR (Print)	**SIGNATURE**	**PHONE NO.**	**DATE**
Dennis Schreiner	[signature]	7845	8-5-96

H. REMARKS

2 ☒ PREVIOUS EXTENSIONS 1

8/21/96

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature) [signature]

DATE 8/29/96

REQUE

[179]

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)

PCAQR NO. 96-551

PAGE

EXTENSION REQUEST

A. ASSIGNMENT Coordinate Root Cause

☐ CONTINUED

B. ORGANIZATION	DUE DATE
QC	6-3-96

C. PCAQR CATEGORY	EXTENSION REQUESTED UNTIL (See NG-NA-702 Attachment 4)
2	7-22-96

D. REASON FOR EXTENSION REQUEST

More time is needed in addressing the cause + corrective actions needed for this problem

See attached action plan

☐ CONTINUED

REQUESTOR

E. JUSTIFICATION FOR DELAY (ADDRESS EFFECT ON HARDWARE, PROGRAMS, AND ACTIVITIES)

this condition is not reportable + engineering has stated that there is NO NON-CONFORMANCE WRT the RV head

☐ CONTINUED

F. REQUESTOR (Print)	SIGNATURE	PHONE NO.	DATE
G. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
Dennis W. Schaefer	[signature]	7845	6-10-96

H. REMARKS

1. ☐ PREVIOUS EXTENSIONS

not used. See action Plan

[signature] 6/11/96

☐ CONTINUED

I. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	DATE
N/A	

[180]

NRC001-1664

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 17
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

RAS C-158

ED 7032.10

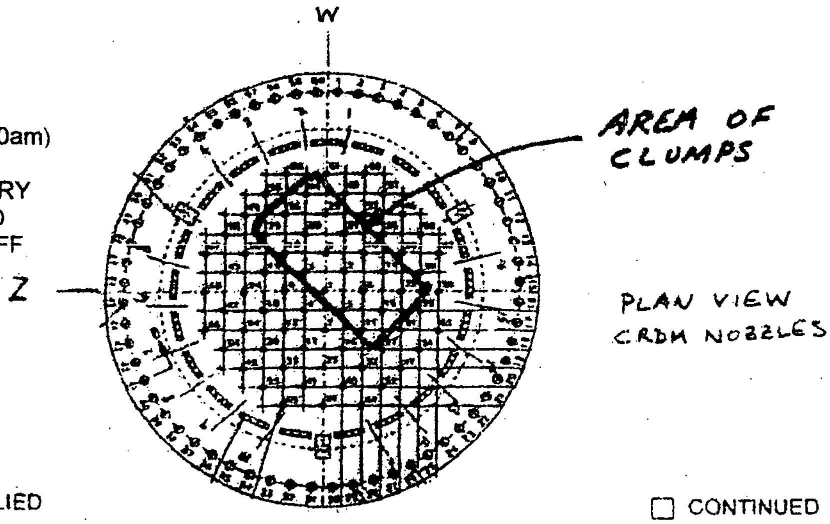
POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR) — PCAQR NO. 1998

PART 1 INITIATION

INITIATOR

A. ISSUE, OBSERVATION, OR CONCERN
Video inspection (4/24/98) of the area where the CRDM nozzles enter the re[…] size clumps of Boric Acid. These clumps were within the area identified by th[…] not present, a light dusting of Boric Acid was found covering the surface area[…]

DOCKETED
USNRC
September 9, 2009 (11:00am)
OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF



☐ Q-HOLD TAG APPLIED ☐ CONTINUED

EQUIPMENT IDENTIFICATION NO. (IF KNOWN)
T1

B. INITIATOR (Print)	SIGNATURE	ORGANIZATION	PHONE NO	MAIL STOP	DATE
Peter J. Mainhardt	[signature]	SYME	8272	1056	4/25/98

SUPERVISOR

C. PCAQR SUBJECT
Reactor Vessel Head Inspection Results

D. SUPERVISOR (Print)	SIGNATURE	ORGANIZATION	PHONE NO	MAIL STOP	DATE
GLENN R. McINTYRE	[signature]	SYME	8165	1056	4.25.98

PART 2 ☐ N/A STATUS AND REPORTABILITY

SHIFT SUPERVISOR

A. MODE / POWER	B. REPORTABILITY	C. OPERABLE
6/0	☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A	☒ YES ☐ NO ☐ NON T.S.

D. IMMEDIATE ACTION TAKEN
None, Plant/Design Engineering will evaluate per procedure.
RCS leakage remained within Tech Spec limits for all of the 11th Operating Cycle.

☐ CONTINUED

E. SHIFT SUPERVISOR (Signature)	DATE	TIME
[signature]	~~4/25/98 1500~~ 4/25/98	1500

S14K-00002

TEMPLATE = SECY 028 DS02

NRC025-1016

ED 7692-10

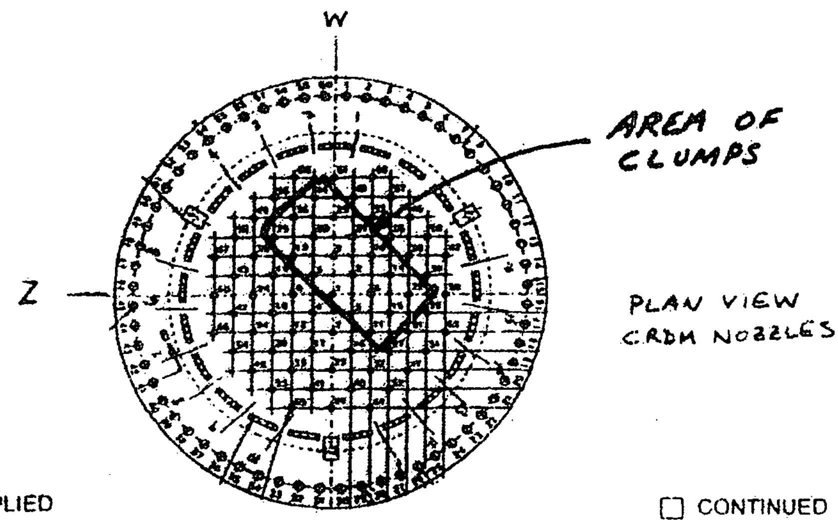
POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 1998-0767	PAGE 1

PART 1 — **INITIATION**

INITIATOR

A. ISSUE, OBSERVATION, OR CONCERN

Video inspection (4/24/98) of the area where the CRDM nozzles enter the reactor vessel head indicated several "fist" size clumps of Boric Acid. These clumps were within the area identified by the plan view herein. Where clumps were not present, a light dusting of Boric Acid was found covering the surface area of the vessel head



☐ Q-HOLD TAG APPLIED ☐ CONTINUED

EQUIPMENT IDENTIFICATION NO. (IF KNOWN)
T1

B. INITIATOR (Print)	SIGNATURE	ORGANIZATION	PHONE NO	MAIL STOP	DATE
Peter J. Mainhardt	[signature]	SYME	8272	1056	4/25/98

SUPERVISOR

C. PCAQR SUBJECT
Reactor Vessel Head Inspection Results

D. SUPERVISOR (Print)	SIGNATURE	ORGANIZATION	PHONE NO	MAIL STOP	DATE
GLENN R. McINTYRE	[signature]	SYME	8165	1056	4 25 98

PART 2 ☐ N/A — **STATUS AND REPORTABILITY**

SHIFT SUPERVISOR

A. MODE / POWER	B. REPORTABILITY	C. OPERABLE
6/0	☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A	☒ YES ☐ NO ☐ NON T.S.

D. IMMEDIATE ACTION TAKEN

None, Plant / Design Engineering will evaluate per procedure. RCS leakage remained within Tech Spec limits for all of the 11th Operating Cycle.

☐ CONTINUED

E. SHIFT SUPERVISOR (Signature)	DATE	TIME
[signature]	~~1500~~ 4/25/98	1500

S14K-00002

NRC025-1016

ED 7032P

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO 98-0767	PAGE 2

PART 4A (CATEGORY 3) INITIAL ASSESSMENT

A. ORGANIZATION: D E M S

DUE DATE: 7/20/98

B. APPARENT CAUSE

The video tape of the reactor vessel head inspection (4/24/98) through the weep holes was reviewed. It showed that most of the head area ~~is~~ was covered with an uneven layer of boric acid along with some large lumps of boric acid

☒ CONTINUED

C. SYSTEM CAPABLE OF PERFORMING SPECIFIED FUNCTION

☐ N/A - Non Tech Spec and not a potentially reportable system.
☒ Yes
☐ No - Initiate new PCAQR if Part 2 is Operable / Yes or Non T.S

See attached

D. 10CFR PART 21 REPORTABLE ☒ NO ☐ YES - NOTIFY REGULATORY AFFAIRS

E. SPECIFIC ACTION NECESSARY TO CORRECT CONDITION (Restoration)

Remove boric acid deposits from the reactor vessel head

REC

JUL 1 1998

☐ CONTINUED

F. WORK COMPLETION DOCUMENTS

The boric acid deposits were removed from the head. The work is documented/captured in a videotape dated 5/4/98.

Note: PCAQR 96-0551 recorded the similar concerns during 10RFO. The root cause evaluation and CATPR for PCAQR 96-0551 is in progress. PCAQR 98-0767 can be closed once the root cause & CATPR for PCAQR 96-0551 are complete.

☐ CONTINUED

G. ASME CODE ☐ NO ☒ YES, ANI REVIEW [signature] 7-16-98 ENGINEERING

H. EVALUATOR (Print)	SIGNATURE	PHONE NO.	DATE
P.K. GOYAL	[signature]	7351	7/16/98
I. SUPERVISOR (Print)	SIGNATURE	PHONE NO.	DATE
T.S. SWIM	[signature]	7799	7/16/98
J. CONCURRENCE - PCAQR REVIEW BOARD CHAIRMAN (Signature)	[signature]		DATE 8/6/98

EVALUATOR

S14K-00003

POTENTIAL CONDITION
ADVERSE TO QUALITY REPORT CONTINUATION SHEET

PCAQR NO. 98-0767 | PAGE 3 OF

PART	ITEM	CONTINUATION
4A	B	The color of the layer and the lumps varied from rust brown to white. The rust or brown color is an indication of the old boric acid deposits. The above tape also showed white streaks on the OD of CRDM housing. This indicates leaking CRDM flanges. It appears that the leaking CRDM flanges contributed to the deposit of boric acid layer and lumps.
4A	C	The reactor vessel head was cleaned as best as we can (The cleaning is recorded on video tape dated 5/5/98). The visual inspection did not show any significant pitting of the head surface. Based on engineering judgement the head thickness (6 5/8") will not be adversely impacted by very slight pitting. Also there were ~~some~~ slight boron deposits left on the head after the cleaning. These deposits will not create any corrosion since the head temperature is ≥ 550°F. This is based on the result of boric acid corrosion test performed by B&WOG (B&W document #51-1229638-1). The testing showed almost no corrosion occurred at temperature greater than 550°F and the highest corrosion rate for carbon steel occurred at approximately 300°F, at the interface where the dried boric acid crystals were re-wetted by the leakage. Lower levels of corrosion occurred at temp. approaching 500°F. The only time the higher corrosion rate can be encountered is during shutdown and heatup when the temperature of head will be well below 550°F. Since this duration is very short no impact on reactor vessel head is anticipated. Thus RCS pressure boundary is not impacted and the RV head will continue to perform its intended function.

[signature] 7/16/98

RECE[IVED]
JUL 1[?] 1998
PCA[...]

☐ CONTINUED

S14K-00004

NRC025-1018

ED 7032-5

POTENTIAL CONDITION ADVERSE TO QUALITY REPORT (PCAQR)	PCAQR NO. 98-761	PAGE 4

PART 7 — VERIFICATION

	YES	NO	N / A
A. APPROVED REMEDIAL ACTION HAS BEEN INITIATED AS DESCRIBED	✓		
B. APPROVED CORRECTIVE ACTION TO PREVENT RECURRENCE HAS BEEN INITIATED AS DESCRIBED			✓
C. Q-HOLD TAGS HAVE BEEN REMOVED			✓
D. CONDITIONAL RELEASE HAS BEEN RESOLVED			✓
E. STOP WORK ACTION HAS BEEN WITHDRAWN			✓

F. DOCUMENTS REVIEWED

☐ CONTINUED

G. COMMENTS

☐ CONTINUED

H. QA REVIEW OF AUDIT / SURVEILLANCE PCAQR N/A			DATE
I. REGULATORY AFFAIRS REVIEW OF REPORTABLE PCAQR N/A			DATE
J. EVALUATOR (Print) [illegible]	SIGNATURE [signature]	PHONE NO. [illegible]	DATE 8/10/98
K. NSI MANAGER CONCURRENCE OF UNACCEPTABLE PCAQR N/A			DATE

NUCLEAR SAFETY AND INSPECTION

S14K-00005

RAS C-159

[CON]DITION REPORT
[]2-1

NO. 2000-1037	PAGE 1 OF 7

EVENT DATE	EVENT TIME	DISCOVERY DATE	DISCOVERY TIME	REFERENCE DOCUMENTS/ASSET NUMBER
		4/6/2000	0330	

CONDITION DESCRIPTION

Inspection of the Reactor Head indicated accummulation of boron in the area of the CRD nozzle penetrations through the head. Borion accummulation was also discovered on the top of the thermal insulation under the CRD flanges Boron accummulated on the top of the thermal insulation resulted from the CRD leakage. The CRD leakage issues are discussed in CR 2000-0782

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 18
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

☒ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
[An]drew Siemaszko	Andrew Siemaszko	04/17/2000	SYME	7341	1056

SUPERVISOR

PLANT OPERATIONS REVIEW ☒ YES ☐ NO

COMMENTS This CR should be sent to SYME for resolution.
This CR will address the effect of the boron on the head.
CR 2000-0782 will address the hardware issue of leaking flanges.

RECOMMENDED CATEGORY
ROUTINE

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
G R McIntyre	[signature]	4-18-00	SYME	8165	1056

PLANT OPERATIONS

REPORTABILITY	OPERABILITY
☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A	☐ YES ☐ NO ☒ N/A

IMMEDIATE ACTIONS TAKEN OR NEEDED / COMMENTS

AS NOTED ABOVE

cc: Initiator
CR Files
~~Nuclear Records Management~~
QIP

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	TIME
PLOEGER R	T. Ploeger	4/18/00	0353

[2354]

TEMPLATE = SECY-028 DS 02

NRC001-0438

RAS C-159

[COND]ITION REPORT
[NOP-LP-2]342-1

NO. 2000-1037

PAGE 1 OF 7

EVENT DATE	EVENT TIME	DISCOVERY DATE	DISCOVERY TIME	REFERENCE DOCUMENTS/ASSET NUMBER
		4/6/2000	0330	

CONDITION DESCRIPTION

Inspection of the Reactor Head indicated accummulation of boron in the area of the CRD nozzle penetrations through the head. Boron accummulation was also discovered on the top of the thermal insulation under the CRD flanges. Boron accummulated on the top of the thermal insulation resulted from the CRD leakage. The CRD leakage issues are discussed in CR 2000-0782

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 18
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

DOCKETED
USNRC

September 9, 2009 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

☒ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
Andrew Siemaszko	Andrew Siemaszko	04/17/2000	SYME	7341	1056

SUPERVISOR

PLANT OPERATIONS REVIEW ☒ YES ☐ NO

COMMENTS This CR should be sent to SYME for resolution.
This CR will address the effects of the boron on the head.
CR 2000-0782 will address the hardware issue of leaking flanges.

RECOMMENDED CATEGORY
ROUTINE

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
G R McIntyre		4-18-00	SYME	8165	1056

SHIFT OPERATIONS

REPORTABILITY	OPERABILITY
☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A	☐ YES ☐ NO ☒ N/A

IMMEDIATE ACTIONS TAKEN OR NEEDED / COMMENTS

AS NOTED ABOVE

cc: Initiator
CR Files
~~Nuclear Records Management~~
QIP

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	TIME
PLOEGER	T Ploeger	4/18/00	0353

[2354]

TEMPLATE = SECY-028 DS 02

NRC001-0438

CONDITION REPORT
ED 8348A-1

NO. 2000-1037	Page 2 of 7

OWNER	CATEGORY	DUE DATE	OPERATING EXPERIENCE REPORT
SYME	ROUTINE	7/17/00	☐ EVALUATE ☐ INITIATE

CAUSE DETERMINATION

☐ CATPR ☒ APPARENT ☐ ROOT CAUSE ☐ MULTI DISC ROOT CAUSE ☐ ERB

☐ EXPERIENCE REVIEW ☐ EXTENT OF CONDITION ☐ POTENTIAL MRFF ☐ OTHER REVIEW REQUIRED

MRC COMMENTS

MODE 4 RESTRAINT - Complete all actions necessary to restore equipment to allow the Mode change. When all actions are complete, document on a Cause/Action Sheet (ED8342B) and provide a copy of the CR to Quality Programs.

Rmd

SUPERVISOR ASSIGNED DUE DATE:

☒ CONTINUED

10 CFR PART 21? ☒ NO ☐ YES	IF YES, DATE	SYSTEM CAPABLE OF PERFORMING SPECIFIED FUNCTION? ☐ YES ☐ NO ☐ N/A

	SIGNATURE	DATE
ANI REVIEW REQUIRED? ☒ NO ☐ YES IF YES, ANI SIGNATURE REQUIRED		
PREPARER (Print) ANDREW SIEMASZKO	Andrew Siemaszko	5/1/2000
SUPERVISOR APPROVAL (Print) Glenn McIntyre		5-1-00
MANAGER APPROVAL (Print) ☐ N/A		
DIRECTOR APPROVAL (Print) ☐ N/A		
VICE PRESIDENT APPROVAL (Print) ☐ N/A		
SRB APPROVAL (Print) ☐ OE REQUIRED ☐ N/A		
ERB APPROVAL (Print) ☐ N/A		

[2355]

NRC001-0439

CONDITION REPORT ED 83428	NO. 2000-1037	Page 3 of 7

See attached response.

☐ Continued

[2356]

CR 2000-1037
4 of 7

Problem statement.
Large deposits of boron have accumulated on the top of the insulation and on the Reactor Vessel Head.

Event description.

Initial Reactor Vessel Head inspection conducted on 4/5/2000 revealed an accumulation of boron on the Southeast Reactor head flange between the head and the studs. Boron deposits were "lava like" and originate from the "mouse holes" and CRD flanges.

Framatome completed the CRD Video Inspection at 1400 on 4/6/2000. Ed Chimahusky and Andrew Siemaszko were present during the inspection. The inspection is documented on the VHS video cartridge. James R. Harris from Framatome Technologies, Ed Chimahusky, and Andrew Siemaszko examined the results of the inspection.

Five leaking Control Rod Drives were identified at locations: F10, D10, C11, F8, and G9. Main source of leakage can be associated with F10 drive. Positive evidence exists that drives F8, D10 and C11 have limited gasket leakage. This condition can propagate at any time and therefore these drives are considered as leaking. [There are no boron deposits on the vertical faces of the flange of G9 drive. The bottom of the flange of G9 drive is inaccessible for inspection due to the boron buildup on the reactor head insulation, not allowing full camera insertion. Since the boron is evident only under the flange and not on the vertical surfaces, there is a high probability that G9 is a leaking CRD.]

Based on the available information, System Engineering recommended replacement of gaskets or repairs for Control Rod Drives located at F10, D10, C11, F8, and G9 as necessary.

Other supporting information.

Review of industry experience indicates that this type of CRD leakage has been identified numerous times in the nuclear industry. Since the leakage is unwanted, the most typical approach to resolve the problem is to replace gaskets and machine flange faces as required.

CR 2000-0994 was issued to evaluate the F10 CRD flange condition.

Video inspection of the F10 flange indicated presence of small pitting on the outer gasket area. The pitting was located on the outer race of the outer gasket. There was no evidence of erosion noted. The remaining surface of the flange was smooth and no evidence of leakage was noted. Due to the external location of the pitting the possibility of gasket leaks was eliminated. Ron Pillow recommended that the condition of the flange is acceptable for gasket replacement without any need of flange machining. This Framatome Technologies evaluation was supported by the Davis Besse System Engineering recommendation not to machine the F10 flange surface.

The apparent cause of the pitting is associated with the manufacturing process and/or the flange casting process. Reasonable assurance exists that the F10 flange was pitted during initial installation.

Conclusions.

System Engineering and Framatome Technologies recommend that the F10 flange be accepted for use in the condition "as found". Other then normally performed lapping process no additional machining is recommended.

CR 2000-0995 was issued to evaluate the D10 CRD flange condition.

Video inspection of the D10 flange indicated presence of large pitting on the inner and outer gasket areas. The pitting was located mainly on the outer race of the inner gasket. The outer gasket area indicated leakage path and flange material loss. There was evidence of erosion noted. Ron Pillow recommended that the condition of the flange is unacceptable for gasket replacement and recommended the D10 flange machining. This Framatome Technologies evaluation was supported by the Davis Besse System Engineering recommendation to machine the D10 flange surface.

The apparent cause of the pitting is associated with the random loss of the CRD flange bolt(s) tension. This resulted with a small steam leak that propagated with time. This type of leak is well recognized and common in the nuclear industry. It is also difficult to predict. Since the cost of preventative actions to verify the elongation of the CRD flange studs are extremely high these actions are typically not recommended.

Conclusions.

System Engineering and Framatome Technologies recommend that the D10 flange be machined and new gasket installed.

The D10 flange has been machined and new gasket installed.

Nuclear Regulatory Commission (NRC) issued Generic Letter 97-01 to holders of operating licenses for pressurized water reactors (PWR's). The letter requires licensee to maintain a program for ensuring a timely inspection of the control rod drive mechanism (CRDM) and other vessel closure head penetrations. The program is required due to degradation of the CRDM nozzles caused by Primary Water Stress Corrosion Cracking process. In order to perform required inspections the nozzles as well as the penetrations must be free of boron deposits. Once the head is free from the boron, new boric acid deposits may be easily noted and remedial actions taken.

CR 2000-1037
6 of 7

Apparent cause.
The apparent cause is CRD flange leakage as identified in CR 2000-0782.

Remedial Actions.
Accumulated boron deposited between the reactor head and the thermal insulation was removed during the cleaning process performed under W.O. 00-001846-000. No boric acid induced damage to the head surface was noted during the subsequent inspection.

Overview of the cleaning effort.
There are two areas requiring cleaning, the area above the insulation and the area below the insulation on the top of the reactor vessel head. The area above the insulation is accessible through the ventilation duct openings located approximately seven feet above the head flange. This area will not require cleaning since it was vacuumed and cleaned by FTI during the F10 and the D10 CRD gasket replacement effort. The area below the insulation on the top of the reactor vessel head will be accessible via the weep holes (other name is mouse holes). The cleaning media will be pressurized de-mineralized water heated to approximately 175 °F. Water will be sprayed on the boron deposits through the ventilation duct openings and through the weep holes. One weep hole will be used to drain the liquid out of the head to the plastic drums. The remaining weep holes will be blocked with a plastic tape. The plastic drums will be located outside of the head stand area at the base of the water shield tanks. Two-inch diameter corrugated plastic hose will provide means of transporting the liquid from the weep hole to the plastic drums. Accumulated liquid will be disposed off as directed by Radiation Protection (RP) personnel. The estimated volume of water used will be between 100 and 600 gallons. Some boron deposits are hardened and soaking time may be required.
Major challenges of the cleaning effort will be associated with spill protection. Recently installed inner and outer Reactor Vessel Head gaskets can not become soaked with the boric acid solution. To protect the gaskets, a number of protective measures will be taken.

- All but one weep hole will be blocked with the plastic cover. In the event the water is escaping from the covered weep hole the cleaning effort will be stopped and spill contained.
- All stud holes will be covered with plastic covers and secured with black tape. Should the liquid escape from the weep hole it will float toward the edge of the head and drip down on the floor surface. It is not likely that the liquid would continue its flow under the flange for approximately 30 inches to reach the gaskets.
- The spray and drain process will be coordinated such that when a spill is noted the spraying operation will be stopped immediately. Only small amount of water will be used at a time.

Another challenge of the cleaning effort will be associated with the protection of the CRDM motors. To prevent water damage to the motors the only area where water will be permitted and sprayed is located between the flange plane and the top of the insulation. The spray operator will be briefed about the need to control the spray and not to create any splashing. The operator will be briefed not to spray any water on the motor assemblies. Motor assemblies are sealed and are not easily impregnable with water.

ALARA considerations will include time/distance principle. The cleaning effort will mainly consist of preparation work. The cleaning effort is scheduled to last approximately 4 hours. The equipment operator will minimize his stay time in the "shine" area while spraying. If feasible a mirror will be utilized to inspect the results of spray at the ventilation duct openings area. After initial cleaning a video inspection will be performed by Framatome Technologies. Should additional cleaning be required the process will be repeated until most boric acid deposits are removed or as directed by RP.

CONDITION REPORT ED 8342B	NO. 2000-1037	Page of

CAUSE/ACTIONS

Removal From Mode Restraint List:

CR2000-0782 addressed the concern of boron on the Reactor Vessel Head. This CR was written for boron on the CRD nozzles on the head, but the review performed under CR2000-0782 encompassed this area. No separate review or evaluation is necessary. The Reactor Vessel Head will be cleaned of all boron deposits following completion of CRD flange repairs by FTI. The cleaning is scheduled and will occur prior to the head is moved from the head stand. No evaluation is needed to support a Mode 4 entry, therefore this CR can be removed from the Mode 4 restraint list.

David C. Geisen

David C. Geisen

4/27/00

☐ Continued

RAS C-160

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 19
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

CONDITION REPORT
ED 8342-1

NO. 2000-0782

EVENT DATE	EVENT TIME	DISCOVERY DATE	DISCOVERY TIME
		4/6/00	0330

CONDITION DESCRIPTION

Inspection of the Reactor flange indicated Boric Acid leakage from the weep h record). The leakage is red/brown in color. The leakage is worst on the east si the weep holes is approx 1.5 iches thick on the side of the head and pooled on top of the flange. The leakage evident from the weep holes appears to be a dried stream in every case. The leakage on the flange are small flakes of Boric Acid that has spalled off from the top of the flow streams and from some of the clumps within the weep holes. The total estimated quantity of leakage through the weep holes and resting on the flange is approx. 15 gallons. All leakage appears to be dry. A very small quantity (approx. 0.25 pint) and run down the side of the flange and onto the floor. Preliminary inspection of the head through the weep holes indicates clumps of Boric Acid are present on the east and south sides. The north and west sides have very little Boric Acid accumulation from the weep holes. The flange studs/nuts do not appear to be affected.

DOCKETED
USNRC

September 9, 2009 (11:00am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
Peter J. Mainhardt	Peter J. Mainhardt	4/6/00	SYME	8272	1056

PLANT OPERATIONS REVIEW ☑ YES ☐ NO

COMMENTS

RECOMMENDED CATEGORY

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
PLOEGER	T. Ploeger	4/6/00	OPS	7501	2103

REPORTABILITY ☐ 1 HR ☐ 4 HR ☐ 24 HR ☑ N/A

OPERABILITY ☑ YES ☐ NO ☐ N/A

IMMEDIATE ACTIONS TAKEN OR NEEDED / COMMENTS

NOTIFIED BACC COORDINATOR PER STEP 6.3.5 OF NG-EN-00324, BA CORROSION CONTROL.

FURTHER EVALUATION REQUIRED AFTER DETAILED INSPECTION DELINEATED IN STEP 6.4.1 OF NG-EN-00324 IS PERFORMED

POSSIBLE MODE 4 RESTRAINT

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	TIME
PLOEGER 2	T. Ploeger	4/6/00	0530

cc: Initiator
CR Files
~~Records Management~~

QIP

A-4

[1772]

AJS OI 0032

TEMPLATE = SECY-028 DS 02

NRC031 0042

CONDITION REPORT
ED 8042-1

NO. 2000-0782	PAGE 1 OF 4

EVENT DATE	EVENT TIME	DISCOVERY DATE	DISCOVERY TIME	REFERENCE DOCUMENTS/ASSET NUMBER
		4/6/00	0330	

CONDITION DESCRIPTION

Inspection of the Reactor flange indicated Boric Acid leakage from the weep holes (see attached pictures and inspection record). The leakage is red/brown in color. The leakage is worst on the east side weep holes. The worst leakage from one of the weep holes is approx 1.5 iches thick on the side of the head and pooled on top of the flange. The leakage evident from the weep holes appears to be a dried stream in every case. The leakage on the flange are small flakes of Boric Acid that has spalled off from the top of the flow streams and from some of the clumps within the weep holes. The total estimated quantity of leakage through the weep holes and resting on the flange is approx. 15 gallons. All leakage appears to be dry. A very small quantity (approx. 0.25 pint) and run down the side of the flange and onto the floor. Preliminary inspection of the head through the weep holes indicates clumps of Boric Acid are present on the east and south sides. The north and west sides have very little Boric Acid accumulation from the weep holes. The flange studs/nuts do not appear to be affected.

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
Peter J. Mainhardt	Peter J. Mainhardt	4/6/00	SYME	8272	1056

PLANT OPERATIONS REVIEW ☒ YES ☐ NO

COMMENTS

RECOMMENDED CATEGORY

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	ORGANIZATION	TELEPHONE NO.	MAIL LOCATION
PLOEGER	T. Ploeger	4/6/00	OPS	7501	2103

REPORTABILITY ☐ 1 HR ☐ 4 HR ☐ 24 HR ☒ N/A

OPERABILITY ☒ YES ☐ NO ☐ N/A

IMMEDIATE ACTIONS TAKEN OR NEEDED / COMMENTS

NOTIFIED BACC COORDINATOR PER STEP 6.3.5 OF NG-EN-00324, BA CORROSION CONTROL.

FURTHER EVALUATION REQUIRED AFTER DETAILED INSPECTION DELINEATED IN STEP 6.4.1 OF NG-EN-00324 IS PERFORMED

POSSIBLE MODE 4 RESTRAINT

☐ CONTINUED

NAME (Print)	SIGNATURE	DATE	TIME
PLOEGER 2	T. Ploeger	4/6/00	0530

cc: Initiator
CR Files
[illegible] Records Management

QIP

A-4

[1772]

AJS OI 0032

CONDITION REPORT
SD 8542A-1

NO. 2000-0732

OWNER	CATEGORY	DUE DATE	OPERATING EXPERIENCE REPORT
SYME	ROUTINE	6/6/00	☒ EVALUATE ☐ INITIATE

	CAUSE DETERMINATION			
☐ CATPR	☒ APPARENT	☐ ROOT CAUSE	☐ MULTI-DISC. ROOT CAUSE	☐ ERB

☐ EXPERIENCE REVIEW	☐ EXTENT OF CONDITION	☐ POTENTIAL MRFF	☐ OTHER REVIEW REQUIRED

NRC COMMENTS

MODE 4 RESTRAINT

Evaluate if an Operating Experience (OE) Report is appropriate and provide a justification/response either way. Please reference NG-NA-00305, Step 6.7, Operating Experience Assessment Program and the Operating Experience Reference Guide. Contact Dennis Snyder if you need assistance.

KMN

SUPERVISOR ASSIGNED DUE DATE:		☒ CONTINUED	
10 CFR PART 21? ☒ NO ☐ YES	IF YES, DATE	SYSTEM CAPABLE OF PERFORMING SPECIFIED FUNCTION? ☒ YES ☐ NO ☐ N/A	
ANI REVIEW REQUIRED? ☒ NO ☐ YES IF YES, ANI SIGNATURE REQUIRED		SIGNATURE	DATE
PREPARER (Print) ANDREW SIEMASZKO		SIGNATURE Andrew Siemaszko	DATE 4/14/2000
SUPERVISOR APPROVAL (Print) Glenn McIntyre		SIGNATURE	DATE 4-27-00
MANAGER APPROVAL (Print) ☐ N/A		SIGNATURE	DATE
DIRECTOR APPROVAL (Print) ☐ N/A		SIGNATURE	DATE
VICE PRESIDENT APPROVAL (Print) ☐ N/A		SIGNATURE	DATE
SRB APPROVAL (Print) ☐ OE REQUIRED ☐ N/A		SIGNATURE	DATE
ERB APPROVAL (Print) ☐ N/A		SIGNATURE	DATE

AJS OI 0033 [1773]

Response to CR 2000-0782

Action Item 1 of this CR is issued to. "Evaluate if an Operating Experience (OE) Report is appropriate and provide a justification/response either way. Reference NG-NA-00305, Step 6.7, Operating Experience Assessment Program and the Operating Experience Guide.

Event description

Initial Reactor Vessel Head inspection conducted on 4/5/2000 revealed an accumulation of boron on the Southeast Reactor head flange between the head and the studs. Boron deposits were "lava like" and originating from the "mouse holes" and CRD flanges.

Framatome completed the CRD Video Inspection at 1400 on 4/6/2000. Ed Chimahusky and Andrew Siemaszko were present during the Inspection. The inspection is documented on the VHS video cartridge. James R. Harris from Framatome Technologies, Ed Chimahusky, and Andrew Siemaszko examined the results of the inspection.

Five leaking Control Rod Drives were identified at locations: F10, D10, C11, F8, and G9. Main source of leakage can be associated with F10 drive. Positive evidence exists that drives F8, D10 and C11 have limited gasket leakage. This condition can propagate at any time and therefore these drives are considered as leaking. There are no boron deposits on the vertical faces of the flange of G9 drive. The bottom of the flange of G9 drive is inaccessible for inspection due to the boron buildup on the reactor head insulation, not allowing full camera insertion. Since the boron is evident only under the flange and not on the vertical surfaces, there is a high probability that G9 is a leaking CRD.

Based on the available information, System Engineering recommends replacement of gaskets or repairs for Control Rod Drives located at F10, D10, C11, F8, and G9 as necessary.

In addition should the examination of flanges (for the above listed drives) indicate steam cuts, System Engineering recommends machining of the flange faces as necessary to ensure acceptable Control Rod Drives performance.
Flange D10 should be machined by FTI.

The proposed sequence of repair is to start with F10 CRD as the prime suspect of the leak, then follow with D-10 as the CRD identified during 11RFO as a suspect of leakage. The remaining Control Rod Drives C11, F8, and G9 can be worked in order convenient to Framatome Technologies. These three leaks identified were significantly smaller in comparison with the leaks observed on F10, and D10 CRDs Control Rod Drives. System Engineering recommends the sequence to be C11, F8, and G9.

Sequence of CRD work is —F10—D10—C11—F8—G9—
This sequence is provided for efficiency of work and is not a requirement.

CATS Followup Item 1 written to complete replacement of gaskets and machining of Flange D10.

☒ Continued

AJS OI 0034 [1774]

Operability determination.

Review of industry experiences indicates that this type of CRD leakage has been identified numerous times in the nuclear industry. Since the leakage is unwanted, most typical approach to resolve the problem is to replace gaskets and machine flange faces as required. In some occasions the leak is monitored and not repaired at the time of discovery. Based on the quantity of the leak found at Davis Besse on 4/5/200 during 12RFO and Control Rod Drives performance during operating cycle 12 all Control Rod Drive mechanisms are operable. No reasonable assurance exists that the leak will not propagate. The estimated leak rate from the flanges is inconsequential for the RCS inventory. Total unidentified RCS leakage was maintained at approximately 0.3 GPM during most of the operating cycle. 12. This is well below the Technical Specifications limit of 1 GPM. Numerous small RCS valves packing leaks were identified during the recent Reactor Building Mode 3 walkdown on 4/1/2000. Control Rod Drives flange leakage was a small contributor to the overall unidentified leak rate of 0.3 GPM.

Operating Experience

Operating Experience Assessment Program procedure NG-NA-00305 was reviewed to identify the need of issuing the OED. Step 6.7 directs Davis Besse personnel to submit the OED to the Operating Experience Coordinator for dissemination upon discovery of an event or a condition that would be of use to the industry.

Flange leaks originating from the Control Rod Drive (CRD) flanges have been identified around 1980. Various attempts were made by Framatome Technologies, ABB, and others to reduce CRD leakage. Initially used asbestos gaskets were replaced with the flexitalic type gaskets. Davis Besse's CRD flexitalic gaskets were replaced with the graphite gaskets in approximately 1992. The size and type of the leak seen at Davis Besse is not unusual. Ron Pillow from Framatome Technologies discussed this issue with Andrew Siemaszko. Ron agreed that no new conditions could be presented to the industry at this time and supported System Engineering position not to issue Operating Experience to the industry. Should the results of CRD leakage investigation revile any new conditions an evaluation will be performed and the Operating Experience will be issued as required.

☐ Continued

BORIC ACID CORROSION CONTROL INSPECTION CHECKLIST

(For documentation purposes only – refer to NG-EN-00324 for procedural requirements)

INITIAL INSPECTION

COMPONENT REACTOR
LOCATION HEAD FLANGE
DATE 4/6/00 MDT/WO#/CR

Amount, thickness, density of boron / Area of component affected:

BA Deposit	Minor	Moderate	Substantial	Dry / Active	Color
Packing				Dry / Active	
Packing gland				Dry / Active	
Yoke arms				Dry / Active	
Bonnet Studs				Dry / Active	
Bonnet Nuts				Dry / Active	
Bonnet				Dry / Active	
Body				Dry / Active	
Piping				Dry / Active	
Other				Dry / Active	Red/Brown

Reason for classification: Heavy Leakage from head weepholes

Identify all other components affected: N/A or list Head, Flange

Component internals affected or area not visible: N/A or list Head, CRD Tubes

Is leakage active? YES NO
Estimate of leak rate:

Recommended method to stop leakage: REWORK

Corrosion present? YES NO
Evidenced by: Red/Brown Deposits

Material Evaluation for affected components/Method of Determination:

Packing Gland	S/S or other	
Yoke Arms	S/S or other	
Bonnet Studs	S/S or other	
Bonnet Nuts	S/S or other	
Bonnet	S/S or other	
Body	S/S or other	
Piping	S/S or other	
Other	S/S or other	Stainless Steel, Carbon Steel.

CR 2000-0782
ATTACHMENT 1
PAGE 1 OF 5

Detailed Inspection Recommended? YES NO
Evaluation can be considered complete if the leakage is deemed Minor, if leakage is on component that is stainless steel or high alloy, or if leakage/corrosion is limited to components that do not affect function.

Basis for recommendation: New Leakage from head which was not evident during 11RFO

RP contacted for clean up? YES NO

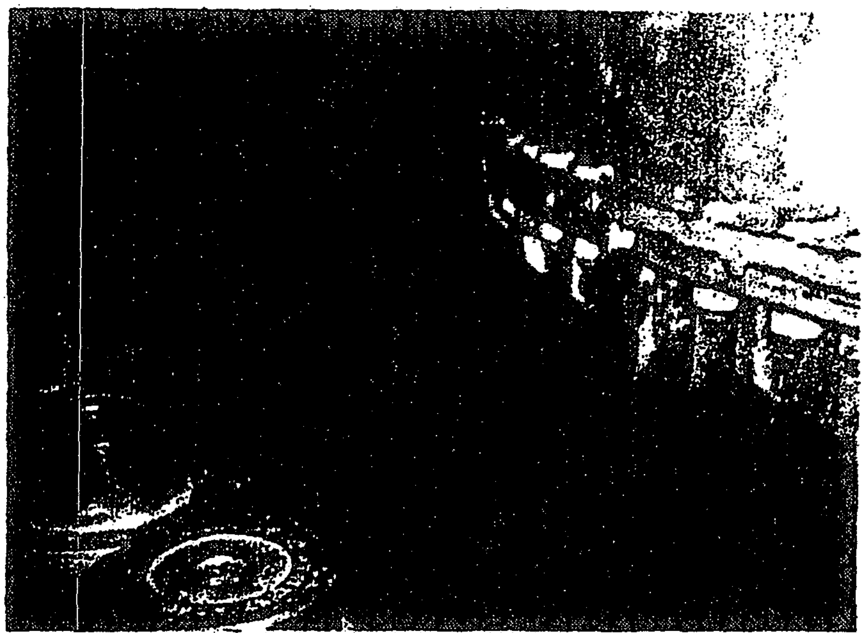
COMPLETED BY [signature] DATE 4/6/00

[1776]

AJS OI 0036

NRC031-0046

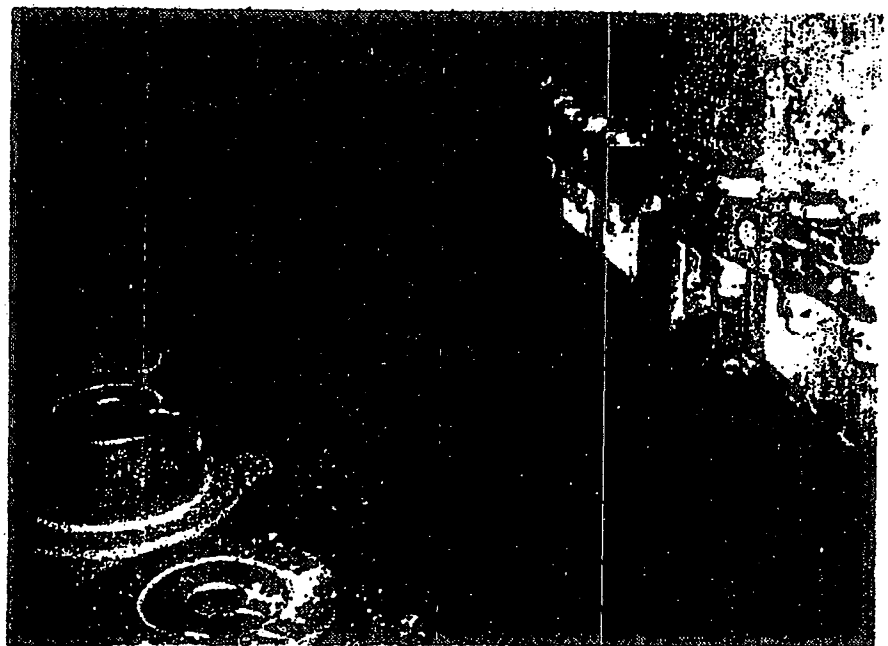
Reactor Vessel Head Boric Acid Leakage
4.6.00



On north side facing east

On north side facing south east

CR 2000-0782
ATTACHMENT 1
PAGE 2 OF 5



On east side facing south

On east side facing south

CR 2000-0782
ATTACHMENT 1
PAGE 3 OF 5

AJS OI 0038 [1778]


On south side facing west

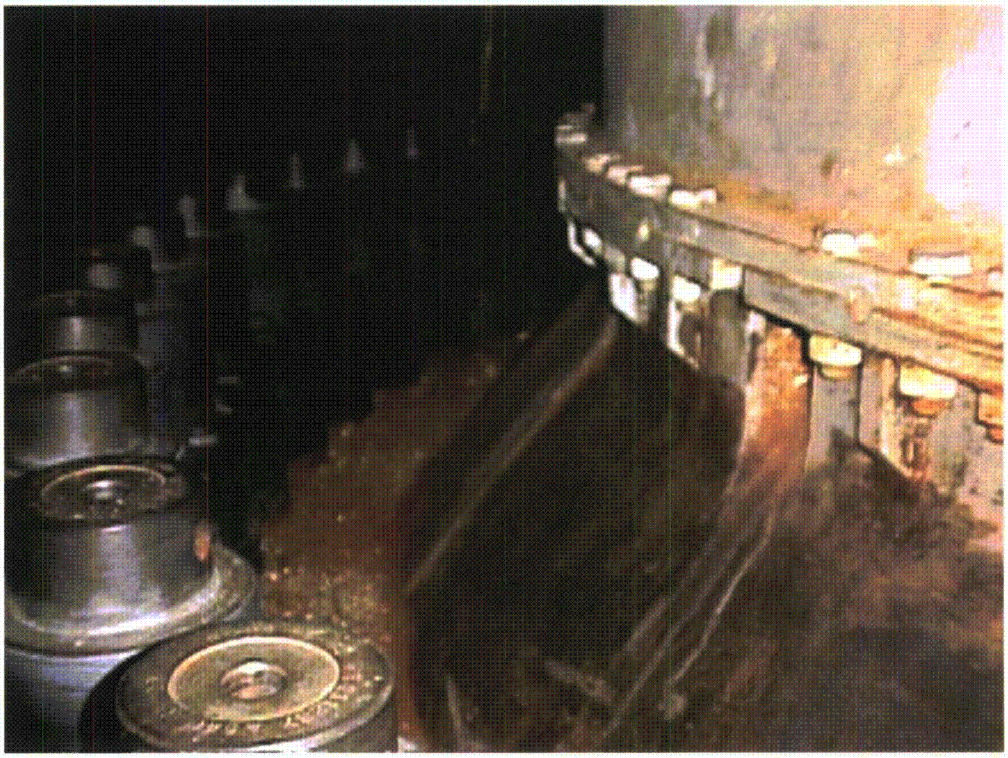

On south side facing west

CR 2000-0782
ATTACHMENT 1
PAGE 4 OF 5


On south side facing north

CR 2000-0782
ATTACHMENT 2
PAGE 5 OF 5

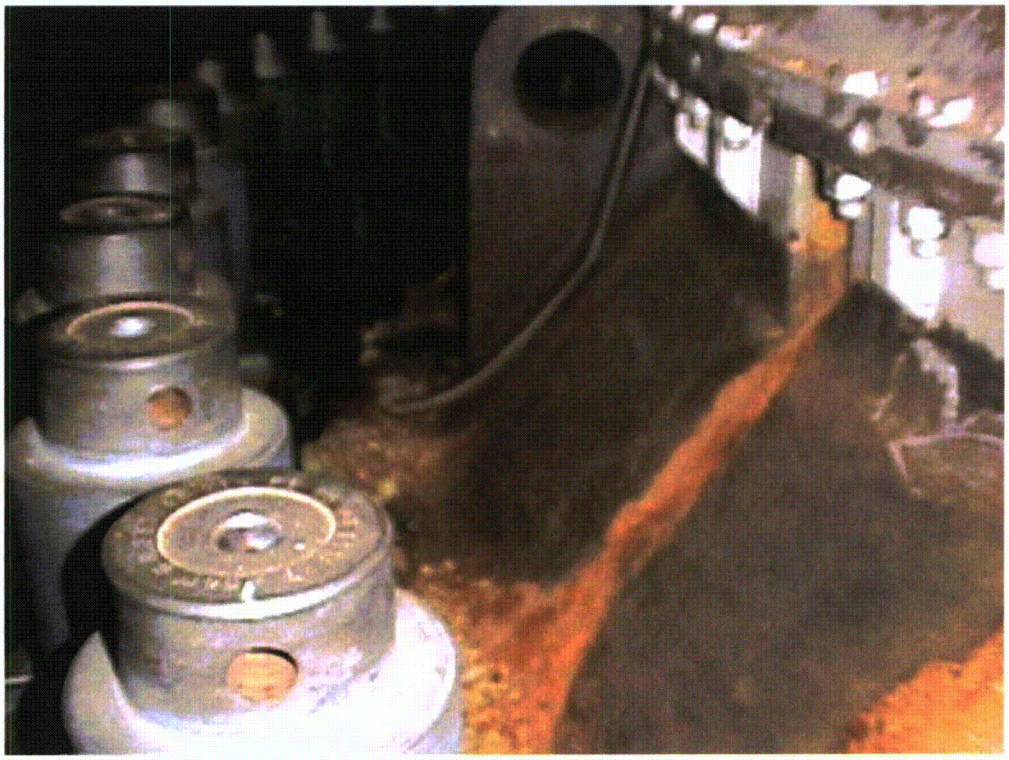
Reactor Vessel Head Boric Acid Leakage
4.6.00


On north side facing east


On north side facing south east


On east side facing south


On east side facing south


On south side facing west


On south side facing west


On south side facing north

RAS C-161

U.S. NRC
In re DAVID GEISEN Staff Exhibit # 20
Docket # 1A-05-052
Date Marked for ID: 12/8, 2008 (Tr. p. 825)
Date Offered in Ev: 12/8, 2008 (Tr. p. 826)
Through Witness/Panel: N/A
Action: ADMITTED REJECTED WITHDRAWN
Date: 12/8, 2008 (Tr. p. 826)

DOCKETED
USNRC
September 9, 2009 (11:00am)
OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

DAVIS-BESSE PLANT

Work Order

T1*RC

Asset: T1*RC REACTOR VESSEL 1-1
Problem Locn: CTMT_213*_565

Action: ROUTINE MAINTENANCE
WO Type:
Clearance: N

Work Class:
Mat. Acct: 4537.4537.IS.NP13A.DBRX.00000.D0
Printed: 25-APR-00 15:51 D2L

Clearance number:
Tech Spec: N
Test Requirements: N
Lead Craft: RADIATION TEST

Quality Class: Q
Environmental Qualification: N
ASME Component: ASMEKI
Repair Tag Number: U0896
Train:

Permission to Commence Work

SS/SM Authorization: N/A DATE

SUPERVISOR DATE

Requested by: ANDREW SIEMASZKO **Phone:** 7341
Planner: DENNIS A LISKA **Phone:** 8338

Problem Description:
LARGE BORON ACCUMULATION WAS NOTED ON THE TOP OF THE RX HEAD AND ON TOP OF THE INSULATION. BORIC ACID CORROSION MAY OCCUR

NO TAGS HUNG (IN CONTAINMENT)
SS/SM APPROVED BY: GARY MELSSEN
FAILURE DATE: 04-21-00
IDA04-21-00

Work Description:
CLEAN BORON ACCUMULATION FROM TOP OF REACTOR HEAD AND ON TOP OF INSULATION.
SEE ANDREW SIEMASZKO (PLANT ENGINEERING), EXT 7341 FOR ADDITIONAL DETAILS.

Work Order Review

Plant Engineering	Andrew Siemaszko	DATE	4/25/00
SRO		DATE	4/25/00
ALARA		DATE	4/25/00
QC Mechanical		DATE	4/26/00
Lead Shop Review		DATE	5/2/00

Special Instructions:
Ensure water does not enter Refueling Canal 4/26/00

Page 1 of 3

30

04/17/2002 DBNPS

TEMPLATE = SECY 028 _ DS02

Exhibit 78
Page 1 of 25 Pages

NRC001-0456

FENOC DAVIS-BESSE PLANT

T1*RC

Work Order 00-001846-000
Subsystem: SUB062-01

Asset: T1*RC REACTOR VESSEL 1-1
Problem Locn: CTMT9_213*_565

Action: ROUTINE MAINTENANCE
WO Type:
Clearance: N

Work Class:
Mat Acct: 4537.4537.IS.NP13A.DBRX.00000.D0
Printed: 25-APR-00 15:51 D2L

Clearance number:
Tech Spec: N
Test Requirements: N
Lead Craft: RADIATION TEST

Quality Class: Q
Environmental Qualification: N
ASME Component: ASMEXI
Repair Tag Number: U0896
Train:

Permission to Commence Work

SS/SM Authorization N/A DATE

SUPERVISOR DATE

Requested by: ANDREW SIEMASZKO Phone: 7341
Planner: DENNIS A LISKA Phone: 8338

Problem Description:
LARGE BORON ACCUMULATION WAS NOTED ON THE TOP OF THE RX HEAD AND ON TOP OF THE INSULATION. BORIC ACID CORROSION MAY OCCUR

NO TAGS HUNG (IN CONTAINMENT)
SS/SM APPROVED BY: GARY MELSSEN
FAILURE DATE: 04-21-00
JDA04-21-00

Work Description:
CLEAN BORON ACCUMULATION FROM TOP OF REACTOR HEAD AND ON TOP OF INSULATION.
SEE ANDREW SIEMASZKO (PLANT ENGINEERING), EXT 7341 FOR ADDITIONAL DETAILS.

Work Order Review

Plant Engineering	Andrew Siemaszko	DATE	4/25/00
SRO		DATE	4/25/00
ALARA		DATE	4/25/00
QC Mechanical		DATE	4/26/00
Lead Shop Review		DATE	5/2/00

Special Instructions:
Ensure water does not enter Refueling Canal. 4/25/00

04/17/2002 DBNPS

Exhibit 78
Page 1 of 25 Pages NRC001-0456

FEARNS DAVIS-BESSE PLANT

T1*RC

Work Order 00-001846-000
Subsystem SUB062-01

Permits

RWP
Transient Combustible

Steps

Craft	Crew Size	Crew Name	Hrs
1 RADIATION TEST	4		10

CLEAN BORON ACCUMULATION FROM TOP OF REACTOR HEAD AND ON TOP OF INSULATION.

SEE ANDREW SIEMASZKO (PLANT ENGINEERING), EXT 7341 FOR ADDITIONAL DETAILS.

1) RAISE LEAD BLANKETS AS REQUIRED TO PROVIDE ACCESS TO WEEP HOLES. ALL BLANKETS WILL HAVE TO BE RAISED TO PROVIDE ACCESS 360 DEGREES AROUND HEAD AT WEEP HOLE LEVEL.

2) INSTALL PROTECTIVE COVERING ON REACTOR HEAD BOLT HOLES. THIS IS REQUIRED TO PREVENT WATER RUN OFF FROM DRAINING THROUGH BOLT HOLES.

3) COVER WEEP HOLES AND PROVIDE DRAIN.

4) POWER WASH REACTOR VESSEL HEAD.

5) REMOVE PLASTIC AND PROTECTIVE COVERS.

6) RESTORE LEAD BLANKETS AS DIRECTED BY RP

SIGNATURE: [signature] DATE: 4/25/00

2 MECHANICAL	4		4

REMOVE AND REPLACE LEXAN COVERS ON REACTOR VESSEL HEAD TO FACILITATE CLEANING.

SIGNATURE: N/A. [signature] DATE: 4/25/00

3 MAINTENANCE SERVICES	1		6

IF NECESSARY MANUFACTURE REPLACEMENT LEXAN COVERS

SIGNATURE: N/A [signature] DATE: 4/25/00

Closeout

Lead Shop / MDT Removed [signature] Date: ~~4/25~~ 5/2/00

SS/SM Authorization N/A [initials] 5/… Date:

QC Mechanical N/A 4/25/0… Date:

Planner Review [signature] Date: 5/…

Completion Date: 4/25/00 Completed By: [signature]

04/17/2002 DBNPS

2

NRC001-0457

FEARMS DAVIS-BESSE PLANT

T1*RC

Work Order 00-001846-000

Subsystem: SUB062-01

Notes:

04/17/2002 DBNPS

3

NRC001-0458

00-001876-000

POST-TEST EVALUATION/DESCRIPTION OF WORK PERFORMED

Work performed without deviations.

[illegible]
4/25/00

CONTINUED ON BACK SIDE

FAILURE EVALUATION/DESCRIPTION OF WORK PERFORMED

PERSONNEL RECORD

USE INITIAL OF FIRST AND MIDDLE NAME AND ENTIRE LAST NAME (Print)

NAME	DISCIPLINE	DATE

MATERIAL DISPOSITION

ITEM DESCRIPTION	STOCK CODE NO.	MIT/PO/ PIC TICKET	QUANTITY USED	COMMENTS

5

00-1046-004

Reactor Vessel Head cleaning.

Large deposits of boron have accumulated on the top of the insulation and on the Reactor Vessel Head. Nuclear Regulatory Commission (NRC) issued Generic Letter 97-01 to holders of operating licenses for pressurized water reactors (PWR's). The letter requires to maintain the program for ensuring the timely inspection of the control rod drive mechanism (CRDM) and other vessel closure head penetrations. The program is required due to degradation of the CRDM nozzle caused by Primary Water Stress Corrosion Cracking process. In order to perform required inspections the nozzles as well as the penetrations must be free of boron deposits. Once the head is free from the boron new boron deposits may be easily noted and remedial actions taken.

Background and technical information.
Beginning in 1986, Alloy 6000 CRDM nozzle leaks have been reported

Overview of the cleaning effort.
There are two areas requiring cleaning. The area above the insulation and the area below the insulation on the top of the reactor vessel head. The area above the insulation is accessible through the ventilation duct openings located approximately seven feet above the head flange. Scaffolding (movable platform) will be utilized to gain access to the ventilation duct openings after Lexan covers will be removed. The area below the insulation on the top of the reactor vessel head will be accessible via the weep holes (*other name is mouse holes*). The cleaning media will be pressurized de-mineralized water heated to approximately 175 °F. Water will be sprayed on the boron deposits through the ventilation duct openings and through the weep holes. One weep hole will be used to drain the liquid out of the head to the plastic drums. The remaining weep holes will be blocked with a plastic tape. The plastic drums will be located outside of the head stand area at the base of the water shield tanks. Two inch diameter corrugated plastic hose will provide means of transporting the liquid from the weep hole to the plastic drums. Accumulated liquid will be disposed off as directed by Health Physics and or Decontamination Department personnel. The estimated volume of water used will be between 100 and 600 gallons. Some boron deposits are hardened and soaking time may be required.

00-001846-00

Major challenges of the cleaning effort will be associated with the spill protection. Recently installed inner and outer Reactor Vessel Head gaskets can not become soaked with the boric acid solution To protect the gaskets number of protective measures will be taken.

- All but one weep hole will be blocked with the plastic cover. In the event the water is escaping from the covered weep hole the cleaning effort will be stopped and spill contained.
- All stud holes will be covered with the plastic covers and secured with the black tape. Should the liquid escape from the weep hole it will float toward the edge of the head and drip down on the floor surface. It is not likely that the liquid would continue its flow under the flange for approximately 30 inches to reach the gaskets.
- The spray and drain process will be coordinated such that when the sill is noted the spraying operation is stopped immediately Only small amount of water will be used at a time.

Another challenge of the cleaning effort will be associated with the protection of the CRDM motors. To prevent water damage to the motors the only area where water will be permitted and sprayed is located between the flange plain and the top of the insulation. The spray operator will be briefed about the need to control the spray and not to create any splashing. The operator will be briefed not to spray any water on the motor assemblies. Motor assemblies are sealed and are not easily impregnable with water.

ALARA considerations include time/distance principle. The cleaning effort will mainly consist of preparation work. The cleaning effort is scheduled to last approximately 4 hours. With majority of time devoted to the head area. The dose is significantly lower at the weep hole area in comparison with the ventilation duct openings area. Equipment operator will minimize stay time in the "shine" area while spraying. If feasible a mirror will be utilized to inspect the results of spray at the ventilation duct openings area. After initial cleaning a video inspection will be performed by the Framatome Technologies. Should additional cleaning be required the process will be repeated until most boric acid deposits are removed or as directed by HP.

Work Order instructions.

The following items are required for support of head cleaning effort.

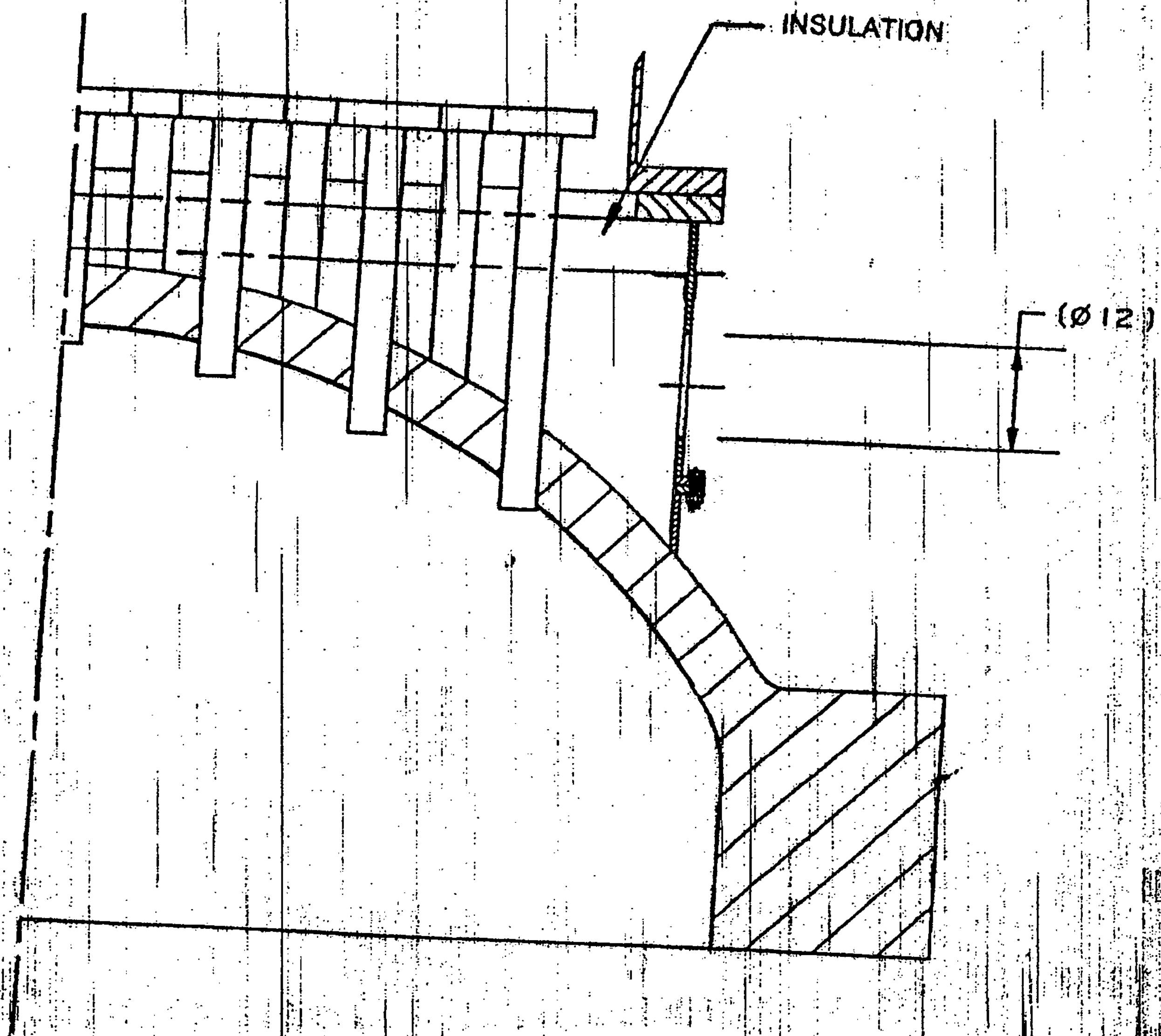
Scaffolding- the scaffold is needed on the North side of the head. The scaffold is needed for wrapping the head with the plastic to block all weep holes. In addition to scaffolding a movable platform will be constructed to enable access to the Lexan covers.

Uncover the weep holes- this can be accomplished by partially rising the bottom portion the lead blankets presently installed on the head. All blankets will need to be raised since plastic tape will be strapped all around the head.

Cover the Reactor Head bolt holes- this can be accomplished by rising the plywood decking and covering the holes with plastic or wrap. Cover each hole

separately by cutting square piece of plastic and tape it to the flange with the black tape. Reinstall the plywood flooring.
[redacted]- Lexan covers are bolted to the ventilation duct openings. The Lexan material is fragile. Special care should be taken during removal and re-installation not to chip any corners and not to overtorque the bolts. This will result in cracks, and covers will have to be replaced. As a precaution, more Lexan sheet material should be ordered in the event that replacement covers are needed. Verify Lexan sheets are available in stores. Materials required to perform the work are: plastic, tarpaulin, black tape, and stainless steel hooks for rising the lead shielding.

SERVICE STRUCTURE SUPPORT INSPECTION OPENINGS



END OF RECORD

04/17/2002 DBNPS

10

NRC001-0465

FENOC DAVIS-BESSE PLANT

T1 RC

Work Order 00-001846-001

Asset [illegible]

Problem Location [illegible]

Action [illegible]

Work Class [illegible]

Mat Acct [illegible]

Printed [illegible]

Quality Class [illegible]

Environmental Qualification

ASME Component

Repair Tag Number

Train

Tech Spec [illegible]

Test Requirements [illegible]

Lead Craft MAINTENANCE SERVICES

Permission to Commence Work

SS/SM Authorization [illegible] DATE [illegible]

SUPERVISOR [illegible] DATE [illegible]/00

Requested by [illegible] Phone [illegible]

Planner [illegible] Phone [illegible]

Problem Description

[illegible]

Work Description

[illegible] SCAFFOLD AS REQUIRED TO SUPPORT BORON REMOVAL

Work Order Review

[illegible] DATE

[illegible] DATE [illegible]

[illegible] Review [illegible] DATE [illegible]/00

Special Instructions

[illegible]

Planning

[illegible]

[illegible]

11

NRC001-0466

FEARMS DAVIS-BESSE PLANT

Work Order 00-001846-001

Shops

Craft	Crew Size	Crew Name
MAINTENANCE SERVICES	6	

... DEGREES AROUND REACTOR HEAD STAND ...
... SHOULD BE BUILT SO TOP ... HEAD ...
... COMPLETION OF ... CLEANING

SIGNATURE [illegible] DATE 4-21-00

RADIATION TEST 2

... SURVEYS AS REQUIRED ...

SIGNATURE [illegible] DATE 4/21/00

Notes

System 24-APR-00 ... HISTORY

Closeout

[illegible] 4/24/00

Completed By

12

NRC001-0467