December 3, 2002

Mr. Lew W. Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 - RESPONSE TO NUCLEAR REGULATORY COMMISSION (NRC) BULLETIN 2001-01, "CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES" (TAC NO. MB2626)

Dear Mr. Myers:

By letter dated December 4, 2001, the NRC staff accepted FirstEnergy Nuclear Operating Company's (FENOC's) response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," and your proposal to operate Davis-Besse Nuclear Power Station Unit 1 (Davis-Besse) until February 16, 2002, rather than December 31, 2001, as requested by the bulletin, before performing reactor vessel head penetration nozzle inspections. The NRC staff's acceptance of your proposed operating schedule followed a substantial dialogue between the NRC and FENOC regarding the technical adequacy of your bulletin response. However, the NRC staff's December 4, 2001, letter did not provide the NRC staff's rationale for accepting your justification for continued operation of the Davis-Besse unit. Rather, the letter stated that the NRC staff's rationale would be documented in separate correspondence. Unfortunately, the NRC staff failed to complete the documentation of its safety rationale once the December 4, 2001, letter had been issued. This oversight was identified by the staff and is discussed in the staff's lessons learned report. In order to fulfill that NRC staff commitment, and in response to requests from other external stakeholders to have the NRC staff's evaluation provided to the public, this letter transmits the NRC's staff's evaluation of your NRC Bulletin 2001-01 response.

On August 3, 2001, the NRC staff issued Bulletin 2001-01 to the industry requesting that addressees provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities. This included 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 at their respective pressurized-water reactor designed plants. In addition, the Bulletin requested that licensees with plants that were considered to be highly susceptible to VHP stress corrosion cracking and that were not planning to perform inspections prior to December 31, 2001, provide the technical basis for their planned inspection schedules.

You provided your response to the information requested in the Bulletin by letter dated September 4, 2001, as supplemented by letters dated October 17, October 30, November 1, and November 30, 2001. In addition, meetings were held on October 24, November 9, November 14, and November 28, 2001, to discuss your responses. Based on the information provided in your responses and the information available to the NRC staff regarding the industry experience with VHP nozzle cracking at the time of the November 28, 2001, meeting, the NRC staff determined that you had provided sufficient information to justify operation until February 16, 2002, and not your originally requested March 31, 2002, date. The commitments contained in your letter dated November 30, 2001, which docketed commitments made verbally on November 28, 2001, were integral to that determination. By letter dated December 4, 2001, the NRC staff accepted your proposal to operate Davis-Besse until February 16, 2002.

This letter and the enclosed NRC staff evaluation are not associated with any ongoing NRC activities related to Davis-Besse. The sole purpose of this documentation is to capture the process used by the NRC staff in the fall of 2001 for review of FENOC's response to NRC Bulletin 2001-01 for Davis-Besse. Because the review of your response was more complex and required a greater amount of NRC staff effort than other Bulletin 2001-01 reviews, the enclosed evaluation provides more detail, and is of a different format, than the documentation provided to other licensees.

Sincerely,

/RA/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure: Staff Evaluation

cc w/encl: See next page

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NUCLEAR REGULATORY COMMISSION (NRC) STAFF EVALUATION

RELATED TO NRC BULLETIN 2001-01 RESPONSE

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1.0 INTRODUCTION

In response to discoveries of cracking in Alloy 600 reactor vessel head penetration (VHP) nozzles at Oconee Nuclear Station Unit 3, in February 2001, and at Oconee Nuclear Station Unit 2, in April 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated August 3, 2001. The bulletin required pressurized water reactor (PWR) licensees to provide information that included: plant-specific susceptibility ranking for primary water stress corrosion cracking (PWSCC); plant configuration information; a description and results of VHP nozzle and reactor pressure vessel (RPV) head inspections performed in the previous 4 years, including any repairs that had been conducted; and, a discussion of plans for performing VHP nozzle and RPV head inspections. Although it was not a regulatory requirement, the bulletin established a milestone of December 31, 2001, as a reasonable goal for completion of the inspections by licensees with plants that had experienced cracking or leakage, and licensees for plants with high susceptibility to leakage. These plants were also requested to provide a technical basis if they did not plan to conduct inspections before that date. A specific concern of the bulletin was the potential for circumferential cracking of control rod drive mechanism (CRDM) nozzles, which might lead to separation of the CRDM nozzle resulting in a loss-of-coolant accident (LOCA).

By letter dated September 4, 2001, FirstEnergy Nuclear Operating Company (FENOC), the licensee for Davis-Besse Nuclear Power Station Unit 1 (Davis-Besse), provided a response to Bulletin 2001-01. Davis-Besse was considered to be a high susceptibility plant for PWSCC. However, FENOC did not plan to perform VHP nozzle or RPV head inspections at Davis-Besse until the 13th refueling outage that, at that time, was scheduled to begin in April 2002.

The NRC technical staff determined that FENOC's September 4, 2001, submittal did not provide sufficient information to completely assess the justification for operating Davis-Besse beyond December 31, 2001. Between September 28, and November 30, 2001, the NRC staff engaged FENOC in substantial dialogue via telephone conference calls, meetings, and docketed correspondence on matters related to the Davis-Besse Bulletin 2001-01 response. FENOC provided supplemental information by letters dated October 17, October 30, November 1, and November 30, 2002. Meetings were held on October 24, November 9, November 14, and November 28, 2001. During this time period, the NRC's Office of Nuclear Regulatory Research (RES), Argonne National Laboratory, and internal engineering studies conducted by the Office of Nuclear Reactor Regulation (NRR) staff provided additional information to the NRC staff.

ENCLOSURE

At the November 28, 2001, meeting, FENOC reaffirmed earlier verbal commitments to, among other things: shut down Davis-Besse by February 16, 2002, instead of March 31, 2002, as originally planned; operate at lower reactor coolant system (RCS) hot-leg temperature; maximize availability of redundant safety systems; provide additional training to operators; perform inspections of 100 percent of the VHPs, and; provide other specific additional information. FENOC formally docketed these commitments by letter dated November 30, 2001. Based on the proposed compensatory measures, the additional information provided on the docket by FENOC, and further evaluation of that information by the NRC staff, the NRC staff determined that the risk associated with continued operation of the Davis-Besse plant until February 16, 2002, was acceptably small.

The NRC staff documented its acceptance of FENOC's justification for continued operation of the Davis-Besse unit beyond December 31, 2001, in a letter to FENOC dated December 4, 2001. However, that letter did not provide the NRC staff's rationale for accepting FENOC's justification, but stated it would be provided in separate correspondence. This assessment documents that rationale. As such, it is based only on the information that the NRC staff was aware of at the time the decision was made.

2.0 BACKGROUND

Several provisions of the NRC regulations and plant operating licenses (technical specifications (TSs)) pertain to the issue of VHP nozzle cracking. Title 10 of the *Code of Federal Regulations* (10 CFR), Appendix A, general design criteria (GDC) for nuclear power plants 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 GDCs applicable to VHP nozzle cracking include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the reactor coolant pressure boundary (RCPB) 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. GDC 31 specifies that the RCPB shall be designed with sufficient margin to assure that the probability of rapidly propagating fracture of the RCPB is minimized. The presence of cracked and leaking VHP nozzles is not consistent with either of these GDCs. 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. Although Davis-Besse is not a plant that was designed to meet the GDC, it was designed to similar requirements.

NRC regulations at 10 CFR 50.55a state that American Society of Mechanical Engineers (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 the ability to reliably detect leakage to the RPV head surface for a through-wall crack in a VHP nozzle. 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 outside diameter 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 TSs pertain to the issue of VHP nozzle cracking insofar as they require no through-wall RCPB leakage. For Davis-Besse, TS 3.4.6.2 specifies that, in Modes 1, 2, 3, and 4:

> Reactor Coolant System leakage shall be limited to:
>
> a. No PRESSURE BOUNDARY LEAKAGE,
> b. 1 GPM [gallons per minute] UNIDENTIFIED LEAKAGE,
> c. 150 GPD [gallons per day] primary-to-secondary leakage through the tubes of any one steam generator,
> d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,
> e 10 GPM CONTROLLED LEAKAGE, and
> f. 5 GPM leakage from any Reactor Coolant System Pressure Isolation Valve as specified in Table 3.4.2.

3.0 EVALUATION

Based on FENOC's responses to NRC Bulletin 2001-01, the NRC staff determined that Davis-Besse was in compliance with the applicable portions of the regulations. Specifically, the surveillance requirements of the ASME Code had been met. Although the GDC do not specifically apply to Davis-Besse because the construction permit for Davis-Besse was issued prior to May 21, 1971, the date that the GDC became effective, FENOC provided information in its September 4, 2001, submittal to demonstrate that it had met the intent of the GDC. Nevertheless, the identification of leakage from CRDM nozzles at similar plants, and the possibility that leakage existed from CRDMs at Davis-Besse, led the NRC staff to question whether the TS requirement of "no reactor coolant pressure boundary leakage," was being met. The NRC staff's interpretation of this requirement was that, in the absence of RCPB leakage, licensee action would only be initiated if an unidentified reactor coolant leakage limit of 1 GPM were exceeded. Compliance with a specific TS action statement could not be enforced based on evidence of leakage at other nuclear plants. The NRC staff was not aware of any direct evidence of RCPB leakage at Davis-Besse.

The question before the NRC staff was whether the existing CRDM inspection methods and intervals provided reasonable assurance of adequate protection of the public health and safety. NRC regulations allowed licensees to conduct the inspections using techniques defined in ASME Code without removing insulation or other impediments. In light of inspection findings at Oconee Unit 3 in the spring of 2001, it was clear that such inspections would not be effective in detecting the very small amounts of boric acid that were expected to be deposited by CRDM nozzle leaks.

The visual inspections required by the ASME Code, as embodied in 10 CFR 50.55a, were not sufficient to detect CRDM nozzle cracking similar to that identified at the Oconee plant. In light of this, the adequate protection provision of 10 CFR 50.109 was used as the basis for evaluation of FENOC's justification for continued operation beyond December 31, 2001. Specifically, the NRC staff used a risk-informed approach which was initially published in Regulatory Issue Summary (RIS) 2000-07, "Use of Risk-Informed Decisionmaking in License Amendment Reviews," dated March 28, 2000, and later finalized in RIS 2001-02, "Guidance on Risk-informed Decisionmaking in License Amendment Reviews," dated January 18, 2001. That risk-informed approach had recently been approved by the Commission for use in decisions related to licensing actions. Although this was not a license amendment review, the NRC staff concluded that the logical thought process presented in RIS 2001-02, and later added to Regulatory Guide (RG) 1.174, was applicable to this case. When the NRC invokes adequate protection as the basis for regulatory action, the staff must demonstrate that the current requirements do not provide an adequate level of protection.

The risk-informed logic entails a process of evaluation which can be summarized in three steps. First, the staff must conclude that the situation at issue represents a special circumstance; that is, a circumstance in which the current requirements do not clearly provide an adequate level of safety. A special circumstance might relate to a situation which was not contemplated when the current requirements were put in place. In this instance, the special circumstance was that the visual inspections required by the ASME Code were not sufficient to detect CRDM nozzle cracking.

The second step was to examine the five attributes (safety principles) of risk-informed decisionmaking as articulated in RG 1.174; namely, 1) compliance with current regulations, or alternative approval, 2) defense in depth, 3) maintenance of sufficient safety margins,

4) minimizing changes in risk, and 5) the ability to monitor the effects of a risk-informed decision. In the case of CRDM nozzles, the important consideration was the likelihood that one of the CRDMs would have a circumferential crack resulting in failure and ejection, thereby causing a LOCA. If the likelihood of a LOCA was significant, then there would not only be a loss of defense in depth and a loss of safety margin, but also a significant increase in risk. The guidelines of RG 1.174 do not constitute a definition of "adequate protection." Rather, they provide a set of criteria to be used in the process of evaluating adequate protection.

The third step was to examine the qualitative factors that enter into judgements about adequate protection. For example, one might look at the GDC to determine the fundamental intentions of the regulatory framework. The requirement most applicable to the Davis-Besse case is GDC 14. In evaluating the licensee's justification for continued operation through March 31, 2002, the NRC staff considered that the important technical parameter was the likelihood that one of the CRDMs would have a circumferential crack resulting in failure and CRDM ejection. If such a crack was likely, the guidance of RG 1.174 would not favor continued operation since a high probability of CRDM ejection would represent a loss of defense in depth (attribute 2). That is, there would be a higher probability of losing one of the three major barriers to fission product release; the RCPB. A high probability of failure of a CRDM nozzle would also represent a clear loss of safety margin (attribute 3). Finally, a high LOCA likelihood, coupled with the estimated conditional core damage and containment failure probabilities, would result in risk estimates that would exceed the guidelines of RG 1.174 (attribute 4).

A high likelihood of LOCA would not be consistent with the qualitative factors that were considered as part of step three of the risk-informed process because the intent of GDC 14 would not be met. Conversely, if the likelihood of CRDM nozzle failure were judged to be small, the guidelines of RG 1.174 would be met. Moreover, the intent of GDC 14 would be satisfied by assuring a high degree of reliability of the RCPB. The NRC staff did not consider it necessary that the licensee demonstrate strict conformance with the "extremely low" criteria for the intent of GDC 14 to be met.

Based on an evaluation of the CRDM nozzle cracking observed at the Oconee plant, the physical process that could lead to CRDM nozzle failure was understood to proceed in three steps: 1) a through-wall crack results in reactor coolant leakage to the annulus region between the nozzle and the vessel head, 2) the exposure of the tube outer diameter to reactor coolant leads to the initiation of a circumferential crack, and 3) the circumferential crack grows to a critical size and the nozzle fails.

The NRC staff was aware that other Babcock & Wilcox Co. (B&W) plants had performed inspections of VHP nozzles and identified leaking through-wall cracks. Some cracks were axially oriented and others were circumferentially oriented. Those that were oriented circumferentially were of greater concern because they are more likely to result in a VHP nozzle tube failure. The NRC staff suspected that leaking cracks existed at Davis-Besse, however, there was uncertainty about their orientation and size. Nevertheless, the NRC staff's judgement was that even the most advanced existing cracks, which had been identified at Oconee Unit 3 and other plants, had considerable structural margin to failure. The NRC staff's preliminary technical assessment estimated a safety margin of three above operating pressure for a through-wall crack of 270°, and higher margins for the largest cracks observed at operating plants. The existence of this margin was part of the basis for concluding that a CRDM nozzle failure leading to a LOCA was very unlikely at Davis-Besse.

The probability of through-wall axial cracks was known to be dependent on a number of factors, including the years of operation of the plant and the temperature of the reactor vessel upper head. The combination of these two factors describes a plant's susceptibility to CRDM nozzle cracking. As described in Bulletin 2001-01, a plant's relative susceptibility to CRDM nozzle cracking was evaluated by determining the future operation time, in effective full power years (EFPY), for the plant to progress to an effective degradation condition (a function of operating time and RPV upper head temperature) equivalent to that of Oconee Unit 3 when that plant identified circumferential cracking. In this case, a low "EFPY to Oconee Unit 3" indicated higher susceptibility to CRDM nozzle cracking. Davis-Besse was judged to be a high-susceptibility plant.

Among the B&W plants, Davis-Besse had less temperature-adjusted operating time than Arkansas Nuclear 1 and the three Oconee units, but more than Three Mile Island, Unit 1, and Crystal River, Unit 3. The results of inspections in the spring and fall of 2001, had revealed circumferential cracks at Oconee Units 2 and 3, with two of the Oconee Unit 3 cracks extending through the nozzle wall. The only other circumferential crack was at Crystal River Unit 3, and it was assessed to not extend through-wall. This qualitative comparison of plant characteristics was judged to be an indicator that circumferential through-wall cracks of the same magnitude as Oconee Unit 3 would be less likely at Davis-Besse.

The NRC staff's evaluation of FENOC's response to Bulletin 2001-01 included information on the quality of past CRDM nozzle inspections at Davis-Besse. The licensee provided information regarding visual inspections performed in 1996 and 1998. These inspections were performed under the insulation installed on the RPV head and, therefore, exceeded the requirements of the ASME Code. Their inspectors attested to the completeness and effectiveness of the inspections. In each of those two outages and in an inspection in 2000, a large fraction of the VHP nozzles were inspected, but no single inspection looked at all of the VHP nozzles. The licensee stated that results showed no evidence attributable to CRDM nozzle leakage. The licensee did not report any other degraded condition of the RPV head, and stated that the RPV head had been cleaned during previous outages.

Upon review of the licensee's presentations, and after viewing videotapes provided by the licensee from the prior inspections in 1996 and 1998, the NRC staff concluded that, while the 1996 inspection was a fairly complete visual inspection of the RPV head, the inspection conducted in 1998 was more limited in scope and quality because of the presence of boric acid deposits. The licensee indicated that those deposits were due to CRDM flange leaks and not through-wall leakage of the CRDM nozzles. The inspection conducted in 2000 was considered to be less effective.

In a series of written submittals and presentations to the NRC staff between September 28, and November 30, 2001, the licensee provided analyses of the frequency of a postulated LOCA as a result of CRDM nozzle ejection. These analyses relied on fracture mechanics calculations based on the results of past CRDM nozzle inspections at Davis-Besse. The licensee further calculated the increase in annual core damage frequency and large early release frequency. The increase in LOCA probability, core damage probability and large early release probability were calculated for the period of operation beyond December 31, 2001.

The licensee proposed a set of interim compensatory measures, docketed in a letter dated November 30, 2001, to mitigate the additional risk associated with an increased LOCA frequency. First, FENOC documented moving their shutdown date for the 13th refueling outage forward from March 31, 2002, to February 16, 2002. Second, they committed to lower the operating temperature of the upper RPV head from 605 °F to 598 °F. This reduction in temperature was estimated to represent a very small decrease in the growth of any axial or circumferential cracks that might exist. In addition, the licensee proposed actions to increase the reliability of emergency core cooling systems (ECCS) and, thus, lessen the potential consequences of a CRDM nozzle ejection. These actions included the use of a dedicated operator to perform a number of manual actions outside of the control room, which are required to switch over to the recirculation mode of the ECCS in the event of a LOCA; failure of switch over is a principal contributor to ECCS failure in the Davis-Besse probabilistic risk assessment. Other precautions to maximize the availability of ECCS equipment were also included in the licensee's commitments. The licensee also provided estimates of the changes in the probability of core damage and large early release due to the proposed compensatory actions.

From the "Preliminary Technical Assessment of Reactor Pressure Vessel Head Penetration Nozzle Cracking" prepared by the NRC staff and issued on November 6, 2001, conclusions can be drawn regarding the amount of crack growth that would be expected to occur with additional plant operation until February 16, 2002. From Figure 23 of that assessment, operation at 598 °F for six weeks would not result in appreciable crack growth of existing flaws. In particular, a through-wall flaw that is greater than 240° around the circumference of the nozzle would continue to satisfy the ASME Code margin of three times the design pressure during the continued operation period until February 16, 2002, and any through-wall flaw less than 310° in circumferential length would not grow to failure within this operating period. It was noted that the largest circumferential crack at Oconee Nuclear Station Unit 3, the plant with the highest susceptibility to cracking, was 165°. Thus, it was considered very unlikely for Davis-Besse to have existing through-wall circumferential cracks that would result in nozzle failure and ejection.

The NRC staff reviewed the base-case risk calculations provided in FENOC's November 1, 2001, submittal and in subsequent presentations and performed sensitivity analyses to determine the effect of prior inspections on risk. The NRC staff also considered the licensee's evaluation provided in support of the compensatory measures. The NRC staff estimated that, giving credit only to the inspection performed in 1996, the probability of a CRDM nozzle ejection during the period of operation from December 31, 2001, to February 16, 2002, was in the range of 2E-3 and was an increase in the overall LOCA probability for the plant. The increase in core damage probability and large early release probability were estimated as approximately 5E-6 and 5E-8, respectively.

Based on:

1. the NRC staff's assessment of the structural capability of CRDM tubes and potential growth of circumferential flaws likely to exist;

2. review of operational experience related to CRDM leakage, including the leaking nozzles identified at other B&W plants;

3. the analytical work performed by the RES staff in support of this effort; and,

4. information provided by FENOC regarding past inspections at Davis-Besse;

the NRC staff concluded that the likelihood of a LOCA due to a CRDM nozzle ejection during the period of operation from December 31, 2001, and February 16, 2002, was acceptably small. Additionally, given the conditional core damage and containment failure probabilities for such an event, the NRC staff concluded that the overall risk increase would also be acceptably small.

4.0 CONCLUSION

Because the likelihood of a LOCA was somewhat increased, FENOC proposed a set of interim compensatory measures to partially mitigate the additional risk associated with an increased LOCA frequency. These included shortening the remaining period of operation, lowering the operating temperature of the upper RPV head, and enhancing the reliability of the ECCS.

The NRC staff concluded that the likelihood of a LOCA at Davis-Besse due to CRDM nozzle ejection during the period of operation from December 31, 2001, to February 16, 2002, was acceptably small. The NRC staff also concluded that, given the conditional core damage and containment failure probabilities for such an event, the overall risk increase would be acceptably small and defense in depth was preserved. This conclusion was based on the weight of evidence including: the structural margins to failure in nozzles identified with circumferential cracks at other plants; a comparison of operating time of Davis-Besse with other B&W units; the favorable inspections results from the 1996 outage; and, the NRC staff's review of probabilistic safety assessment calculations submitted by the licensee. The NRC staff further concluded that, while the structural margins of some CRDM nozzles had the potential to be significantly reduced, sufficient margin remained to maintain safety and prevent a LOCA. Thus, the NRC staff concluded that the change in core damage frequency due to the potential for CRDM nozzle ejection was consistent with the guidelines of RG 1.174.