September 5, 2003

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 1 - ISSUANCE OF AMENDMENT (TAC NO. MB8953)

Dear Mr. Myers:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 257 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit 1. The amendment revises the Technical Specifications (TS) in response to your application dated May 14, 2003, as supplemented by letters dated June 16, August 2, August 7, and August 20, 2003.

This amendment revises the TS to allow a one-time exception, only during the Restart Test Plan, to allow entry into Mode 3 of operation without the high-pressure injection pumps being able of taking suction from the low-pressure injection trains when aligned for containment sump recirculation. The exception cannot be used for entry into Mode 2 or Mode 1.

In the May 14, 2003, application, the licensee requested the Nuclear Regulatory Commission (NRC) to review the proposed changes to the TS on an exigent basis. By letter dated June 16, 2003, the licensee stated that based on delays in the restart schedule that the exigent circumstances no longer existed and expedited review of the amendment was not necessary.

During the review of the proposed amendment, the NRC staff found that it needed additional information in order to complete its review. The NRC staff developed questions and forwarded them to the licensee on June 25, 2003. The questions were discussed with the licensee in a meeting at Framatome in Lynchburg, Virginia, during the week of July 14, 2003. Additionally, the NRC staff had a teleconference with the licensee on July 24, 2003, to provide further clarification of the questions. By letter dated July 31, 2003, the NRC issued the final set of questions. By letter dated August 2, 2003, the licensee responded to the request for additional information. Upon review of the licensee's August 2, 2003, letter, the NRC staff identified a concern regarding the execution of the oath and affirmation submitted as part of the response. The NRC staff informed the licensee of the concern by telephone on August 7, 2003. By letter dated August 7, 2003, the NRC stated that, while the NRC staff has not made any conclusion regarding this matter, in order for the NRC to proceed with the review of the requested license amendment, the licensee must address this concern. By letter dated August 7, 2003, the licensee addressed the NRC staff's concern.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Jon B. Hopkins, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Amendment No. 257 to
License No. NPF-3
2. Safety Evaluation

cc w/encls: See next page

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

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cc w/encls: See next page

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*by memo to A. Mendiola dated 8/25/03.
**See previous concurrence
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Davis-Besse Nuclear Power Station, Unit 1

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FIRST ENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-346

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 257
License No. NPF-3

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment by the First Energy Nuclear Operating Company (the licensee) dated May 14, 2003, as supplemented by letters dated June 16, August 2, August 7, and August 20, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-3 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 257 , are hereby incorporated in the license. FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 5, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 257

FACILITY OPERATING LICENSE NO. NPF-3

DOCKET NO. 50-346

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3/4 5-3	3/4 5-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 257 TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By application dated May 14, 2003, as supplemented by letters dated June 16, August 2, August 7, and August 20, 2003, the FirstEnergy Nuclear Operating Company (FENOC) (the licensee) requested an amendment to the Technical Specifications (TS) for the Davis-Besse Nuclear Power Station (DBNPS), Unit 1. The proposed amendment would revise the TS to allow an exception only during the current refueling outage to allow entry into Mode 3 operation without the high-pressure injection (HPI) pumps being capable of taking suction from the low-pressure injection (LPI) trains when aligned for containment sump recirculation. The exception cannot be used for entry into Mode 2 or Mode 1.

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

1.1 Summary of Amendment Request

The proposed amendment would change the Limiting Condition for Operation (LCO) for Section 3.5.2 of the TS. LCO 3.5.2 applies to the emergency core cooling systems (ECCS) for the average reactor coolant system (RCS) temperature, $T_{avg} \geq 280$ °F (Mode 3). In part, it states, “Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of: a. One OPERABLE HPI pump.” The requested amendment adds the following exception to LCO 3.5.2(a):

> An exception applies to the HPI pumps for the purpose of conducting Restart Test Plan inspection activities. This exception is valid during the ongoing 13th Refueling Outage for entries into MODE 3 from MODE 4. Under this exception, neither HPI train is required to be capable of taking suction from the LPI trains when aligned for containment sump recirculation. The HPI trains will otherwise be operable. Operation in MODE 1 or MODE 2 while relying upon the provisions of this exception is prohibited.

The primary purpose of the request is to allow a seven-day test at approximately 2155 psig and 532 °F (Mode 3) to confirm that there are no RCS pressure boundary leaks. The heatup will principally use reactor coolant pump (RCP) heat since the decay heat generation rate is too small to achieve the desired temperature.

1.2 Background Regarding HPI Pump Inoperability

The ECCS includes two independent trains. Each train contains a LPI pump and an HPI pump. In the event of a loss-of-coolant accident (LOCA), the ECCS pumps are automatically started upon receipt of a safety features actuation system signal with suction aligned to the borated water storage tank (BWST). In larger LOCAs, the RCS depressurizes rapidly to a pressure less than the shutoff heat of the LPI pumps, and HPI pump injection is not necessary with the possible exception of a core flood line (CFL) break. In smaller LOCAs, RCS pressure may remain above the shutoff head of the LPI pumps, necessitating injection from an HPI pump to ensure adequate core cooling. For long-term operation following depletion of the BWST, LPI pump suction is switched to the containment emergency sump and HPI pump suction is switched to the LPI discharge piping.

On October 22, 2002, with the reactor defueled, Licensee Event Report (LER) 2003-002 (Reference 1) identified a potential deficiency of the HPI pumps. This deficiency would occur during the recirculation phase of a LOCA and when the HPI pumps are used for post-LOCA boron precipitation control. The HPI pumps may be damaged from debris after the pump suctions are switched over from the BWST to the discharge of the LPI pumps, which are taking suction on the containment emergency sump. The HPI pumps use a process-fluid lubricated hydrostatic radial bearing on the outboard end of the pump shaft. The hydrostatic bearing, an inter-stage bearing, and wear rings may be damaged by debris or particles in the pumped fluid. On the driven end of the pump, an oil-lubricated bearing supports the shaft. The HPI pumps were declared inoperable since the ability to maintain long-term core cooling while taking suction through the original sump screens was in question.

This HPI pump inoperability does not allow FENOC to take the DBNPS into Mode 3 to confirm that there is no RCS pressure boundary leakage. The requested exception, if granted, would allow the Mode 3 test to be conducted. This is the principal topic of this safety evaluation (SE).

This concern only applies when taking suction from the containment emergency sump. Approximately 20 years of experience with HPI pump tests and the general lack of debris in the BWST have demonstrated that the as-built HPI pumps remain operable when suction is from the BWST.

LER 2003-003 (Reference 2) identified a potential condition where RCS pressure may be higher than the discharge pressure from the HPI pump when operating in the piggyback configuration with suction being taken from the containment emergency sump. In this condition, there would have been no flow of ECCS water through the HPI pump and, since ECCS water is used for HPI pump cooling, pump overheating could result. FENOC’s solution for this condition is also addressed in this SE since the condition would render the HPI pumps unavailable, an undesirable condition when considering defense-in-depth.

2.0 REGULATORY EVALUATION

In General Design Criterion (GDC) 35, “Emergency core cooling,” in Appendix A to 10 CFR Part 50, the Commission establishes the design requirement for ECCS by stating that “a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.” GDC 35 continues with requirements for suitable redundancy and the capability to accomplish the system safety function while assuming a single-failure. The staff’s evaluation of the licensee request addresses the licensee’s continued capability to meet the intent of this design requirement with HPI pumps that are inoperable when the ECCS is configured for recirculation from the containment emergency sumps.

In 10 CFR 50.46, the Commission establishes its regulatory requirements for the acceptance criteria for ECCS. 10 CFR 50.46(a)(1)(i) requires that ECCS cooling performance “be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.” 10 CFR 50.46(a)(1)(ii) applies to the DBNPS plant and effectively states that the DBNPS ECCS evaluation model will conform “with the required and acceptable features of appendix K ECCS Evaluation Models.” The staff’s evaluation addresses the licensee’s capability to apply its approved evaluation models to meet the following acceptance requirements of 10 CFR 50.46(b) with HPI pumps that are inoperable when configured for recirculation from the containment emergency sumps:

(1) Peak cladding temperature shall not exceed 2200 °F.

(2) Maximum cladding oxidation shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) Maximum hydrogen generation shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The staff’s evaluation further addresses assurance that the most severe postulated LOCAs are calculated.

In 10 CFR 50.57, the Commission establishes requirements pertinent to issuance of operating licenses. 10 CFR 50.57(a)(3) states that “there is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in” 10 CFR. The staff’s evaluation addresses continued compliance with these requirements.

In 10 CFR Part 100, the Commission establishes requirements pertinent to release of radioactive material. The staff’s evaluation addresses continued compliance with this regulation.

3.0 TECHNICAL EVALUATION

The technical evaluation provided below addresses two broad considerations regarding the requested exception to TS 3.5.2 for purposes of an approximately one week Mode 3 test:

(1) Realistic DBNPS response if a LOCA were to occur during the proposed test, and

(2) Compliance with the requirements identified in Section 2.0, above.

Realistic considerations provide perspective and contribute to ensuring that no conditions are overlooked that may be of potential concern. Realistically, some circumstances are addressed that are not typically addressed during design basis reviews for power operation and these are found not to be of concern. Further, the likelihood of core damage due to a LOCA under these circumstances is shown to be small. Establishing regulatory compliance, as proposed by FENOC, is accomplished by procedures to throttle HPI pump flow to achieve RCS depressurization before initiation of recirculation from the containment emergency sump, thereby avoiding the need for HPI under conditions where HPI operability is not ensured. Again, realistically, the likelihood of initiating such procedures is shown to be small.

3.1 Technical Perspective

Relatively short half-life fission products contribute most of the decay heat during the first few days following shutdown from power operation. During an extended time following power operation, the short half-life fission products decay and the long half life fission productions become the dominant contributors to decay heat. Long half life fission products continue to accumulate during power operation whereas short half life fission products reach an equilibrium between production rate and decay rate in a relatively short time. Thus, the 10 CFR Part 50 Appendix K requirement to assume an infinite irradiation time has limited impact on decay heat generation rate within the first few days following shutdown, but the effect of the assumption increases with time after shutdown as the short half life fission products decay.

The DBNPS nuclear reactor has been shut down since February 16, 2002. The short half life fission products have decayed and only the long half life fission products remain. Further, 76 of 177 fuel assemblies in the DBNPS core have not been irradiated. As a result, two differences are immediately identifiable between the proposed Mode 3 test and traditional considerations of accidents initiating from power operation:

(1) The small inventory of radioactive material means that the existing licensing basis release predictions for accidents that are postulated to initiate during full power operation far exceed releases that would be predicted for DBNPS for similar accidents. Consequently,

no further consideration of release and dose is needed provided no accidents are found where core damage could exceed what is predicted for design basis accidents that initiate during full power operation.

(2) The licensing basis decay heat generation rate required by 10 CFR 50 Appendix K is a factor of five greater than the realistic heat generation rate.

Decay heat generation rate often dominates calculated RCS pressure and RCS inventory following postulated small-break LOCAs that initiate during power operation. During the proposed DBNPS Mode 3 operation, calculated RCS pressure response during many small-break LOCAs is dominated by ECCS injection rate since the decay heat generation rate is low. This difference between power operation and existing DBNPS conditions, in turn, lead to different accident control responses. For example, aspects where the DBNPS response to a potential LOCA during the proposed Mode 3 test is predicted to differ from expected response during power operation includes:

(1) There is no fission heat generation prior to initiation of postulated accidents. Consequently, there is no stored energy in the fuel and the traditional rapid heatup associated with larger LOCAs does not occur.

(2) A realistic estimate of the decay heat generation rate is approximately 250 kW. This will boil about 2 gpm of water in an adiabatic system. The licensing basis rate is five times greater, or enough heat to boil 10.5 gpm in an adiabatic system. Any break that is large enough to be classified as a LOCA will remove the decay heat from the break and decay heat removal does not require supplemental cooling. This condition differs from a LOCA that occurs during power operation, where initial boiloff rates from small LOCAs are several hundred gallons per minute and rates of 100 gpm are typically expected several days after shutdown.

(3) The low decay heat generation rate generally makes the dominant accidents less challenging by resulting in greater margins in safety-related injection flow rates, greater margins in operator response times, greater margin in feedwater capacity, greater margin in heat exchanger capacities, longer heat-up times, and reduced RCS inventory boiloff rates.

(4) The low decay heat generation rate will often result in BWST depletion rates that are less than would be required if a comparable LOCA were to occur from power operation. Sufficient time will often be available to initiate BWST makeup, thus extending the time before switchover to the containment emergency sump would be necessary.

(5) In the proposed test, most postulated LOCAs will cool down the RCS via flow out the break while the core temperature remains at the RCS water temperature. The potential exception is a bottom-of-vessel break. A number of incore monitoring instrument (IMI) tubes would have to break to be of concern. One difference between the response to IMI LOCAs during the proposed test and during power operation is a more rapid core heatup in the power operation case.

(6) If one postulates all water is drained from the reactor vessel (RV) with an RCS temperature of 200 °F, and the assumption is made that no heat is transferred from any

fuel rod, a realistic calculation shows that it would take greater than seven hours for the hottest part of the core to reach a temperature that would jeopardize fuel cladding.

(7) A reactor trip from full power will produce an upset of the electrical transmission grid and a fast bus transfer from the auxiliary transformer to the startup transformers. Loss of offsite power may result. These upsets will not occur during the proposed Mode 3 test because no electrical power is being produced. The likelihood of a loss of offsite power immediately proceeding or during conditions requiring ECCS are consequently less during the proposed Mode 3 test when contrasted to power operation.

(8) Some of the cold leg break LOCAs require effective condensing heat transfer within the RV to enhance depressurization rate. This is more important for some of the conditions that potentially could occur during a LOCA from the proposed Mode 3 test. Therefore, core flood tank (CFT) isolation to eliminate injection of non-condensible gasses is more important than during response to small LOCAs that initiate during power operation.

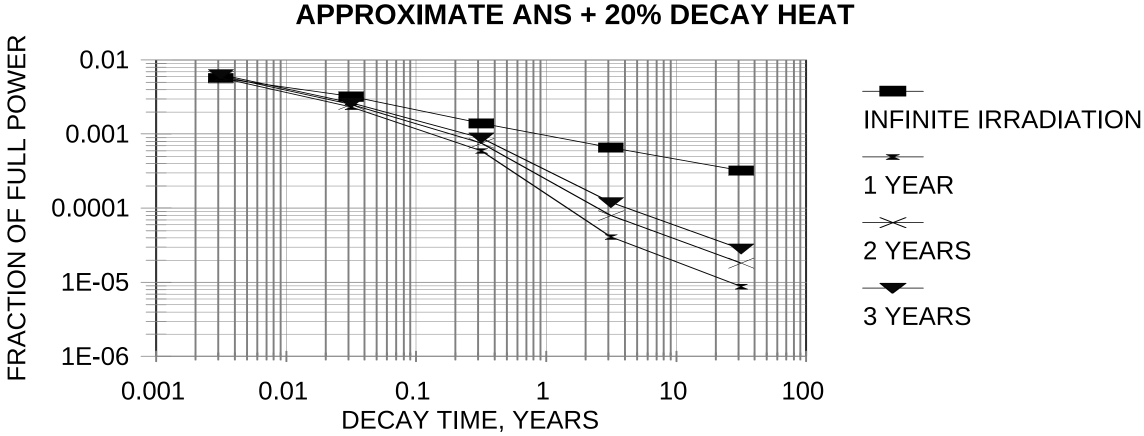
The technical bases for realistic behavior and for compliance with 10 CFR Part 50 Appendix K, and the application of this material to establish regulatory compliance during the proposed Mode 3 test, are examined in the remainder of Section 3, below.

3.2 Heat Generation Rate Considerations

3.2.1 Decay Heat Generation Rate

The requirements of 10 CFR Part 50 Appendix K apply to the LOCA Evaluation Model (EM) for DBNPS. Appendix K requires decay heat generation rate to be calculated using 1.2 times the October 1971 American Nuclear Society (ANS) Standard for an infinite irradiation time. The 1.2 was selected as a reasonable uncertainty bound based upon a perceived uncertainty of 20 percent for the first 1000 seconds and 10 percent thereafter with most of the EM calculations completed within the first 1000 seconds at the time of promulgation of Appendix K. The infinite irradiation time was selected as an upper bound. As seen in the following figure, the selection of infinite irradiation time did not introduce a large penalty in traditional LOCA calculations because decay heat generation rate within a few thousand seconds of LOCA initiation time is a weak function of irradiation time (1000 sec $\equiv$ 3X10^{-5} yrs).

APPROXIMATE ANS + 20% DECAY HEAT



The DBNPS has been shut down since February 16, 2002, and 76 of 177 fuel assemblies have not been irradiated. It was refueled in the spring of 2000 and in the spring of 1998. Thus, the minimum irradiation time for irradiated fuel was about two years and irradiated fuel has been decaying for about 1.5 years. For the conditions applicable to the FENOC amendment request, the Appendix K infinite irradiation time requirement causes calculated decay heat generation rate to be a factor of five higher than would be predicted for realistic irradiation times.

With respect to the staff's review, two decay heat generation rates are considered: (1) the unrealistically high value applicable to the licensing basis requirements; and (2) a more realistic value that bounds plant response. FENOC used decay heat generation rate fractions of 0.00089 and 0.00021, respectively, for a full core of irradiated fuel. FENOC stated that the 0.00089 value corresponded to 394 days post-trip. The plant will have been shut down for more than 530 days when the proposed Mode 3 test is initiated. The staff, using the data for the above figure, calculated 0.00085 for 530 days after power operation. FENOC's licensing basis value of 0.00089 is conservative and is acceptable.

FENOC estimated a realistic irradiation time would result in a 1.2 times ANS decay heat fraction of 0.00021. The staff independently assumed an irradiation time of 2 years and calculated 0.00017. FENOC's estimate is conservative and is acceptable.

3.2.2 Heatup Considerations

FENOC stated that a decay heat fraction of 0.00089 resulted in a core power of 1.44 MWt or 1360 BTU/sec. At one atmosphere pressure, this decay heat would boil water at a rate of (1360 BTU/sec) (60 sec/min) / (970.3 BTU/lb) = 84.1 lbs/min, or (84.1 lbs/min) (0.01672 ft^3/lb) (1728 in^3/ft^3) / (231 in^3/gal) = 10.5 gpm. Thus, if one assumes an adiabatic condition, makeup at 10.5 gpm will keep the core cooled at one atmosphere pressure using licensing-basis assumptions. The staff's realistic decay heat fraction of 0.00017 would result in a boiloff rate of (10.5)(0.00017)/(0.00089) = 2 gpm.

FENOC calculated the adiabatic core hot spot heatup rate to be 0.24 °F/sec assuming the 0.00089 licensing basis value and further assuming there was no water in the RV. To heat from 500 °F to 1400 °F using the licensing basis assumptions would take (900)/[(0.24)(3600)] = 1 hour. The staff estimates a bounding value using realistic assumptions is >(1)(0.00089)/(0.00017) = >5 hours. The staff has audited the FENOC heatup assumptions and calculations, and finds them to be conservative and acceptable.

RCS heat losses are small in comparison to decay heat generation rate for LOCAs that initiate from full power conditions. Therefore, heat loss is neglected when analyzing traditional design basis LOCAs. Consistent with this approach, FENOC has not credited heat loss in its amendment request. However, under conditions that would exist during the proposed Mode 3 test, heat loss may have more influence on potential conditions. Consequently, FENOC provided the following design heat loss information to allow development of insights on how heat loss potentially affects RCS behavior:

Component	Loss Rate, BTU/hr	Loss Rate, kW
28” ID Cold Legs (4)	65,000	19.05
Pressurizer surge line	11,500	3.37
36” ID Hot legs (2)	141,000	41.32
Pressurizer Spray line	12,100	3.55
Steam Generator (SG) exterior (2)	376,000	110.20
Reactor, lower vessel	287,500	84.26
Reactor, head	75,000	21.98
Control Rod Drives	389,000	114.01
Pressurizer (insulated)	66,000	19.34
Pressurizer (uninsulated area)	50,000	14.65
RC Pump insulation loss (4)	100,000	29.31
TOTAL	1,573,100	461.04

Consider, for example, the case of a filled RCS or an RCS with an inventory that will support boiler-condenser heat transport, and assume for both cases that there is no steam generator (SG) cooling and no injection. Heat loss rate, exclusive of the pressurizer, is 420.13 kW. But the staff’s realistic heat loss rate, developed in Section 3.2.1, above, is 250 kW. Assuming a uniform RCS temperature and radiant heat loss behavior, the RCS would reach an equilibrium temperature of about 450 °F with this heat generation rate.

Additional insight may be gained by considering an attempt to heat up the RCS on May 20, 2003, to support a 250 psig pressure test. With the SGs in wet layup, circulation through the decay heat removal (DHR) loop, and insulation removed from many small piping connections to the RCS and from the RCPs, decay heat coolers were bypassed to begin the heatup. RCS temperature increased from approximately 93 °F at an initial rate of about 20 °F per day. By the afternoon of May 22, 2003, RCS temperature had begun to plateau at approximately 120 °F. A small additional increase to approximately 125 °F was attained by spraying the pressurizer which caused an outflow from the pressurizer and effectively used pressurizer heaters to heat the RCS.

Containment water level will also affect RCS heat removal. Following a LOCA, the containment water level is expected to submerge the RV to a level between slightly above the top of the core to just below the hot leg nozzles. Review of insulation details shows that water would enter the mirror insulation, and would be in direct contact with the RV. The annular space between the RV and the insulation would fill to approximately the same level as the water level in the containment. It would not be displaced by steam pressure due to vent paths between numerous insulation panels and the lifting of inspection panels if a small differential pressure were developed due to boiling. This would provide effective heat transfer through boiling. FENOC believes that submergence of the lower part of the RV, by itself, would provide adequate core cooling if required to do so during the planned Mode 3 test. The staff agrees that the FENOC assessment is reasonable. The staff did not credit this cooling mechanism in its review.

In conclusion, the staff finds that the decay heat generation rate is sufficiently low that, in the complete absence of removal of heat by flowing water, RCS heatup will occur at a slow rate and, as illustrated above, the maximum temperature will be substantially below normal operating temperature. Further, as is developed in Section 3.3, below, LOCAs of concern in the licensing basis will have a high enough flow rate to remove all decay heat through the break without causing an RCS heatup or RCS repressurization.

3.3 RCS Pressure Response During Postulated LOCAs

The RCS response to a LOCA from full power operation is different from the response to a LOCA from the proposed Mode 3 conditions. It is important to understand these differences since they strongly affect operator mitigation actions. The differences will be developed in the remainder of Section 3.3 by first considering response to LOCAs that initiate from power operation and then addressing behavior under the proposed Mode 3 conditions.

System response to the following categories of leaks and LOCAs will be described, assuming the operation of one ECCS train:

(0) Leaks that may be mitigated without use of ECCS (< 0.002 ft^2)

(1) Small break (SB) LOCAs that may not interrupt natural circulation with makeup (MU) pump flow (0.002 to 0.005 ft^2)

(2) SBLOCAs that may allow the RCS to repressurize in a saturated condition (0.005 to 0.035 ft^2)

(3) SBLOCAs that allow the RCS pressure to stabilize at approximately the secondary side pressure (0.035 to 0.06 ft^2)

(4) SBLOCAs that depressurize the RCS to the CFT pressure (0.06 to 0.25 ft^2)

(5) SBLOCAs that depressurize the RCS nearly to the containment pressure (0.25 to 0.75 ft^2)

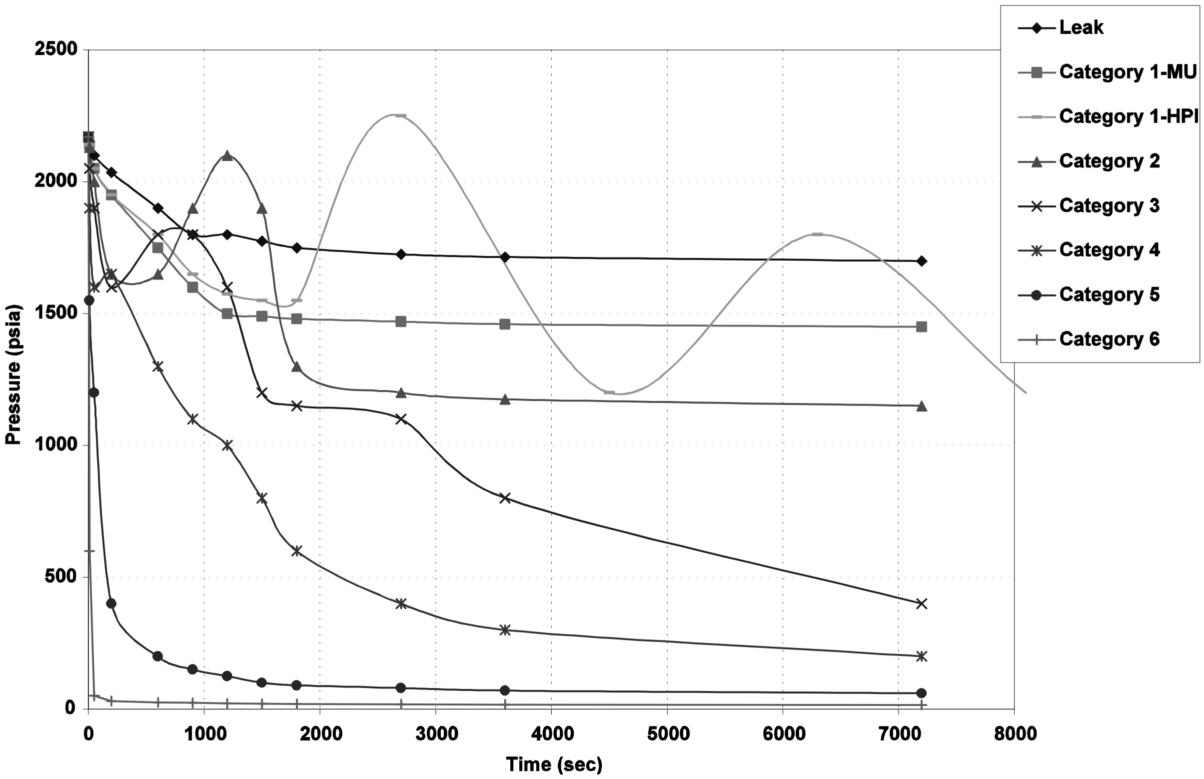
(6) Large break (LB) LOCAs (0.75 to 14.1 ft^2)

RCS pressure behavior for each of these categories for initiation of the accident during power operation is illustrated in Figure 1.

For the DBNPS, MU capacity is considered as the first MU pump in operation followed by start of the second MU pump at 10 minutes after break initiation. If this MU pump combination can maintain an adequate pressurizer level, then it's a leak (category 0), not a LOCA. The predicted maximum break size that corresponds to a leak is a function of break location, makeup and letdown flow rates, the critical flow model used in the analysis, plant equipment available, and operator action. Accordingly, a variety of break areas may be applicable. For DBNPS, a break area less than 0.002 ft^2 is a reasonable representation of breaks that constitute leaks. This would cause a loss of RCS inventory at a rate of about 300 gpm or less with two MU pumps running at normal operating temperature and pressure. As shown in

Figure 1, a leak may be characterized by a small initial decrease in RCS pressure, followed by a steady state condition where injection rate and leak rate are in balance.

Figure 1. Full Power, High Decay Heat Post LOCA Representative Pressure-Time Histories



The DBNPS procedures for RCS leaks instruct the operators to attempt to isolate the leak and to initiate a plant cooldown and depressurization. Based on a BWST volume of 500,000 gallons, more than 24 hours would be available before the BWST would be depleted if the leak could not be isolated. Additionally, the leak rate would decrease as RCS pressure is reduced. In practice, the plant would be cooled down and depressurized to establish DHR operation, or would be placed on LPI recirculation if DHR operation was not possible. Note that this process does not require HPI operation, and therefore, RCS leaks do not require reconsideration of 10 CFR 50.46.

A LOCA is defined as an RCS pressure boundary break that causes loss of coolant at a rate in excess of MU capability. The behavior of the smallest LOCA category, Category 1, is illustrated by two lines in Figure 1, one with MU pump operation and one without. These smallest LOCA break sizes will depressurize slowly and achieve a reactor trip within the first 10 to 20 minutes following initiation of the break. When MU pump flow is credited with some or all of the operator actions identified for leak mitigation, the RCS characteristics will evolve similarly to those of a leak. However, if only redundant safety-related equipment is used with limited credit for operator actions, as is assumed for a design basis calculation, subcooling margin (SCM) may be lost and the RCS inventory loss can result in an interruption of natural circulation. Loss of natural circulation will result in core boiling and RCS repressurization above the HPI shutoff head. When the RCS pressure is above the HPI shutoff head, the RCS liquid inventory and water level will decrease due to the break flow. When the water level descends into the SG tube bundle below the auxiliary feedwater (AFW) injection location, a high-elevation boiler-condenser mode (BCM) of heat transfer is established in the SG. This heat transfer will decrease RCS pressure to below the HPI shutoff head and HPI flow will be reestablished. The RCS will subsequently refill to above the AFW injection location and SG heat transfer will again be interrupted. The RCS will again repressurize above the HPI shutoff head and the cycle will be repeated unless natural circulation of RCS water is regained. Absent operator action, these oscillations could last for days or weeks, with only a slight decrease in the peak pressure as the core decay heat generation rate decreases.

A typical Category 2 break will cause a more rapid RCS depressurization than occurred with a Category 1 break. Steam accumulation in the U-bend region of the hot legs will block natural circulation and interrupt SG heat removal within a few hundred seconds. With a loss of SG cooling, some repressurization will occur due to boiling in the core because the break is too small to remove the decay heat generated in the core plus any remaining sensible heat stored in RCS components. The repressurization is halted when the RCS water level decreases so that cooled SG tubes are exposed to RCS steam and BCM initiates. Then SG cooling restarts and the combination of break and SG heat removal matches or exceeds the rate of heat addition to the RCS fluid. In the example, the RCS pressure then reduces to an equilibrium value where the RCS pressure is greater than the SG pressure, and therefore the RCS temperature is greater than the SG secondary side temperature; a condition necessary for SG heat removal. Note also that this pressure is less than the HPI shutoff head so that HPI flow is established. Further note that MU operation is also beneficial, as was the case for Category 1 breaks.

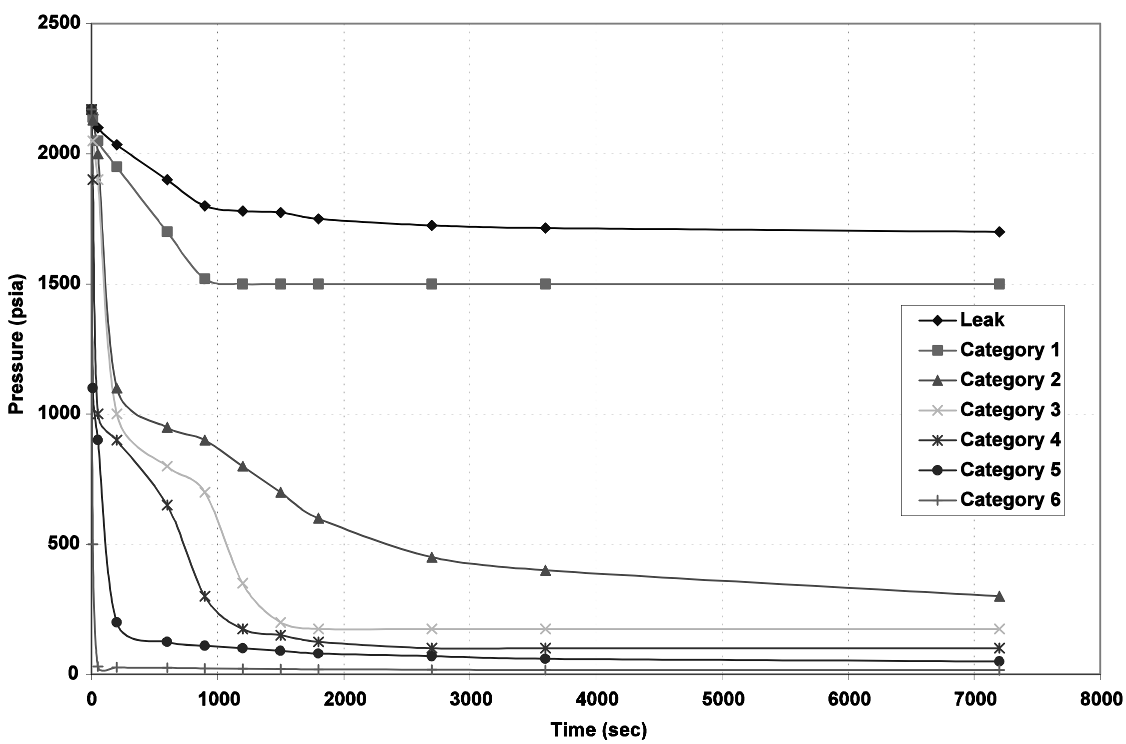
As the break size increases, the break energy discharge to the containment replaces the SG as the primary core heat sink. For Category 3, the break energy discharge will exceed the core decay heat within a few seconds following reactor shutdown. The SG heat transfer via AFW is still important for these break sizes because it can condense RCS steam and enhance the depressurization of the RCS. The condensate combines with the ECCS flow to help limit the ECCS-to-core-boiloff deficit. AFW also cools the secondary side and limits the magnitude of the reverse heat transfer when the break depressurizes the RCS below the secondary side pressure. Typically, it takes 25 to 60 minutes (depending on break size, decay heat power, and SG heat removal) for these break sizes to depressurize the RCS to the CFT pressure. During this time period, the core decay heat boils off the HPI flow that reaches the core and some of the RCS liquid inventory that drains into the RV. The continuous HPI flow delivery to the RV is critical for these break sizes because the RCS liquid inventory available to augment the ECCS is only capable of providing 5 to 10 minutes of core boiloff before the core uncovers.

Category 4 breaks will depressurize the RCS to the CFT pressure within 5 to 25 minutes after break opening. In these breaks, CFT flow halts the core mixture level decrease and initiates RV refill. In addition, the severity of the results somewhat depends on the total HPI flow delivery early during the transient. The AFW fill logic and AFW flow rate are less important on these transients because of the larger break size. Nonetheless, higher AFW flow rates can be beneficial in accelerating the RCS depressurization rate, holding up slightly more liquid in the hot legs and SG tubes, and reducing SG reverse heat transfer.

Category 5 breaks are sufficiently large to depressurize the RCS to approximately the containment pressure as shown in Figure 1. Cold leg breaks in this Category are not large enough to reverse core flow, which would cause the cladding to exceed the critical heat flux upon break initiation. The core is shut down via control rods and cooled during the blowdown transient, which maintains a two-phase mixture that keeps the fuel pin cladding within a few degrees of saturation so long as the mixture level remains above the top of the core. During the rapid depressurization to the CFT pressure, some of these break sizes may cause some core uncovery and cladding heatup. The duration of uncovery is short since CFT flow quickly refills the core and quenches the clad temperature. Depressurization to the LPI initiation pressure will occur within the first 2 to 10 minutes of the LOCA. Consequently, HPI inflow during these first several minutes is of little consequence for core cooling prior to core refill so long as LPI liquid reaches the RV. After the CFTs are empty and the core is refilled, however, LPI and HPI flow provide both diversity of makeup injection sites and sufficient ECCS flow to exceed core boiloff rates. Note, however, that the dependency on HPI is greater in the event of a CFT line break as no LPI flow reaches the RV under design basis assumptions. In this special break configuration, the unbroken CFT and HPI flow must be capable of providing core cooling.

Category 6 breaks will cause the cladding to exceed the critical heat flux upon break initiation. If the break is in a cold leg, core flow may reverse during the blowdown phase. Core cooling during the blowdown and refill phases of the LOCA is by high velocity steam or steam plus liquid droplets. The final cladding quench occurs when the core is reflooded by CFT and LPI flow within minutes after break opening.

Figure 2. Mode 3 Low Decay Heat Post-LOCA Representative Pressure-Time Histories



The RCS behavior is different for the low decay heat Mode 3 conditions that will exist during the proposed Mode 3 test conditions. There is no forcing function for RCS repressurization because any break size relieves the energy generated in the core. Without substantial core heat generation, the secondary side has no potential to repressurize to the main steam safety valve pressures. Note, however, that if the RCPs are running, RCP heat will require SG heat removal. Typical RCS pressure response for the various break categories is shown in Figure 2.

A leak is successfully managed with operator actions and MU flow as described for the full power analyses. Pressure response to a leak is similar to the response for the full power results in Figure 1. The real differences in the pressure trends are observed for the LOCAs. The Category 1 LOCAs have no potential for RCS repressurization unless the ECCS flow is increased, and the RCS pressure consequently evolves to a pressure at which the ECCS injection matches the break flow. The pressure response will be similar if only HPI is credited or if HPI and MU are both credited. The only difference is, if MU is credited, the quasi-steady RCS pressure trend shown in Figure 2 will be higher for the Category 1 breaks.

The Category 2 pressure trends will not plateau at or above the SG secondary side pressure since, when there is little core energy addition, SG heat transfer is not necessary for continued RCS depressurization. Consequently, the RCS will depressurize below the secondary side pressure due to the break energy relief. The Category 3, 4, 5, and 6 pressures also decrease faster because of the minimal core power.

The FENOC pressure response information is generally based on typical licensing basis calculations which assume one ECCS train, a 3.5 percent HPI head reduction, and a similarly generally conservative LPI head curve. When considering potential pressure reduction restrictions where RCS pressure is driven by injection rate into the RCS, maximum flow characteristics should be used. This includes upper bound flow characteristics curves, both ECCS trains, and both MU pumps running. Use of upper bound characteristics will result in equilibration at higher pressure than calculated by FENOC. The effect can be addressed for the conditions corresponding to the proposed Mode 3 test by adjusting the flow area in the FENOC results. For example, the effect of doubling the injection flow rate could be obtained by assuming double the break flow area in Figure 2. In response to a staff question, FENOC stated in Reference 2 that “consideration of the maximum ECCS flows, including two MU pumps, was included in the throttling guidance....Therefore, the report covers both minimum and maximum ECCS flow without exacerbating the severity of the problem or under-predicting the boundary conditions that support a successful demonstration of the 10 CFR 50.46 criteria.” The staff finds that the planned throttling guidance addresses the potential RCS pressure response and the FENOC analysis predictions are acceptable.

3.4 RCS Depressurization

3.4.1 The Potential Need for RCS Depressurization

Figure 2 shows that some of the smaller LOCAs will remain above the LPI shutoff head for an extended time. Although that is not of concern when long-term HPI operation is ensured, this is not the case for the proposed Mode 3 test conditions. During the proposed test, HPI operability is not ensured after the transfer is made to recirculation flow from the containment emergency sump. Consequently, RCS pressure must be ensured to be below the LPI shutoff head prior to the transfer. Depressurization actions may therefore be necessary that are not considered when the full complement of ECCS pumps is operable.

3.4.2 Depressurization Actions

If the LOCA size is large enough that RCS pressure is ensured to fall below the LPI injection pressure before it is necessary to initiate recirculation from the containment emergency sump, and both LPI pumps are injecting water into the RCS at acceptable rates, then HPI flow can be terminated under the emergency operating procedures (EOPs) that have been in place for some time. In this case, 10 CFR 50.46 requirements are met without a need for HPI pump operation when on recirculation.

If the pressure decrease rate is not ensured to allow a sufficient LPI flow rate before switch-over to recirculation, then active steps to accomplish RCS depressurization will be necessary. The first and preferred action will be to control RCPs, SG cooling, and injection rate to accomplish depressurization while maintaining SCM. With SCM controlled by SG heat removal and with pressurizer level within the desired range, generally the MU pumps would be throttled to control pressurizer level while maintaining SCM via SG heat removal and depressurizing via pressurizer spray. Eventually, MU pump operation would be stopped and HPI pump throttling could be initiated. In effect, under these conditions, RCS pressure is being controlled by injection rate since too high an injection rate will result in overfilling the pressurizer; a condition that is prevented by throttling injection rate. These actions are covered in operating procedures that have been in place for some time and the only modification would be one to ensure reaching a sufficiently low pressure prior to initiating switchover to the containment emergency sump.

When running, RCPs are the source of greater than 90 percent of the RCS heat during the proposed Mode 3 test. If SCM cannot be maintained, RCPs are tripped and the source of most of the RCS heat will be removed. For the LOCA sizes of concern, the break flow rate has the capacity to remove all of the decay heat. Consequently, following loss of SCM, SG heat removal would only be needed, in some cases, to remove sensible heat during the early cooldown. Most of the time, the SGs will have little influence on RCS properties following loss of SCM, and the RCS pressure response will be determined by the balance between RCS injection and loss rates. Therefore, if the RCS depressurization rate will not allow effective LPI and will not allow HPI termination before initiating recirculation flow, then the only remaining actions are to increase the effective break size or to throttle injection without SCM.

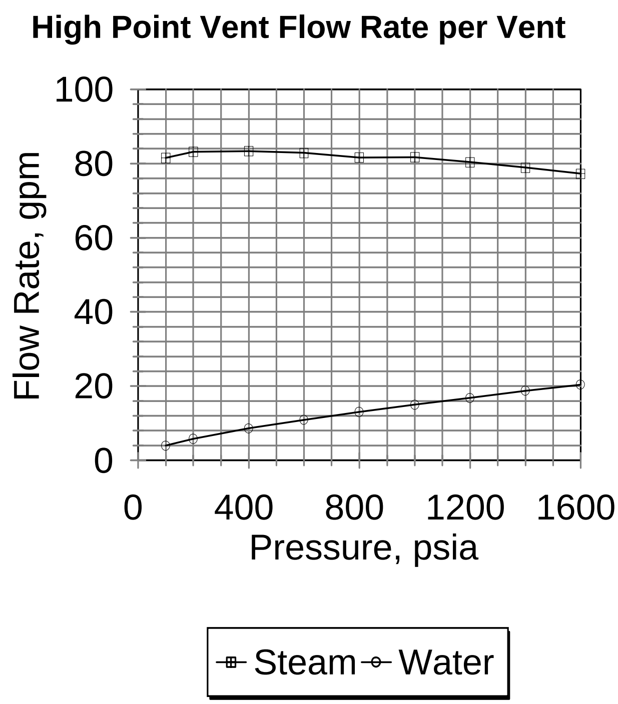
The preferred depressurization action without SCM is to open the pressurizer pilot-operated relief valve (PORV) and the three high point vents. Reference 2 stated that "the PORV area, based on a bore diameter of 1.62 inches, is 0.0143 ft^2. At an RCS pressure of 200 psig, the critical mass flux using Moody with saturated liquid is approximately 3600 $lbm/sec/ft^2$. This corresponds to approximately 425 gpm. The corresponding RCS pressure that results in an LPI flow rate of 425 gpm is approximately 210 psig, which is within a few psi of the LPI pump shutoff head." Flow through the break would provide additional RCS depressurization. Thus, the PORV will cause an initial rapid RCS depressurization that, in conjunction with HPI and MU throttling, will continue past the LPI pump shutoff head and will ensure achieving LPI flow.

The licensee stated that "the PORV was procured and maintained as safety related for its pressure boundary function, however, current documentation does not support environmental qualification under design basis LOCA conditions. Although not currently qualified, the PORV is considered to be highly reliable. This conclusion is based on improvements made to the PORV during 13 refueling outages (RFOs) that included:

- Power cable replacement in containment,
- Solenoid replacement with a safety-related coil,
- Addition of terminations coating system for the solenoid coil,
- Replacement of the power control relay, and
- Rebuilding of the PORV."

Therefore, although the PORV is considered to be reliable, it is not safety grade. Further, there is only one PORV. Thus, the PORV cannot be credited in establishing compliance with 10 CFR 50.46 and GDC 35 because the single-failure criterion is not met.

The high point vent openings are small because each vent has an installed thick-edged orifice with an internal diameter of 0.19 inches (0.0002 ft^2 per vent). However, they do provide an additional depressurization capability.



The high point vents are safety grade, but their effectiveness was not evaluated by FENOC since they are not credited in the 10 CFR 50.46 licensing basis analyses for power operation. To obtain improved insight, the staff examined high point vent effectiveness by using mass flow rate estimates provided by the licensee to generate the volumetric flow rate information shown to the right. As expected, when venting water, the venting capability is relatively low with a maximum flow rate using three vents of about 60 gpm at 1600 psia that ranges to 17 gpm at 200 psig. However, the vent effectiveness changes significantly when venting steam, with a flow rate of 250 gpm at 200 to 400 psig. Note that core cladding damage cannot occur if the core is covered with water, the core cannot become uncovered without exposing the vents to steam, and the boiling rate using licensing basis assumptions is 10.5 gpm. These characteristics indicate that, once LPI is initiated and HPI /MU injection is no longer provided, the vents provide a safety-related capability that ensures continued LPI flow because there is no mechanism that will repressurize the RCS so that LPI is lost. Further, the vents will provide a significant contribution to depressurization actions prior to achieving LPI because:

(1) If a water solid RCS condition exists, RCS pressure will decrease rapidly in response to significant HPI throttling due to flow out of the break, and

(2) If vapor exists in the pressurizer or high point elevations in the hot legs, the high point vents will provide a significant additional volumetric release rate to contribute to RCS depressurization.

A postulated CFL break has the unique characteristic of requiring HPI even when RCS pressure rapidly reduces so that LPI flows at a high rate. This occurs because, if there is a failure of an electrical bus to disable the LPI pump feeding the unbroken CFL, all of the LPI from the operating LPI pump is lost out of the break, and HPI and water from the CFT attached to the unbroken CFL are the only sources of water to the RCS. If this break and postulated failure were to occur, the RV level would be replenished prior to expending the BWST inventory. In the existing licensing basis, long-term cooling is provided by HPI, an option not ensured under the requested exception. FENOC proposes to provide long-term cooling by opening the LPI cross-tie and balancing flow between the two LPI lines prior to the time the BWST is emptied. FENOC points out that this operator action is already procedurally in place, but has not been credited in previous LOCA analyses. It also states that the action is similar to presently credited actions of balancing HPI flow. As a backup for balancing LPI flow in the case of a CFL break, FENOC states that LPI flow can be established through the auxiliary pressurizer spray (APS) line to provide an available redundant flow path. Time to fill the pressurizer and flow rates are discussed in Section 3.10, below. Additional information regarding FENOC's consideration of a CFL break was provided in the licensee's August 2, 2003, letter.

FENOC has predicted, using approved EM methods, that HPI throttling will result in a pressure decrease that will achieve LPI cooling without significant core uncovery. However, throttling HPI without SCM is inconsistent with existing EOPs. The staff's examination of this action is addressed in Sections 3.4.3 - 3.4.6, below. The staff's findings regarding HPI throttling are provided in Section 3.4.7, below.

3.4.3 HPI Pump Throttling

The 1979 Three Mile Island accident led to core damage because operators inappropriately terminated safety injection in the mistaken belief that the RCS was filled with water due to pressurizer level offscale high. The operators did not realize that the core was uncovered and the pressurizer was full because steam was flowing through the surge line into the pressurizer, and no water could return from the pressurizer due to the high steam flow rate. Following this accident, the EOPs for B&W-designed nuclear steam supply systems were rewritten to ensure adequate ECCS water would be provided under all conceivable conditions. This is accomplished by instructing the operators to carefully observe SCM following reactor trip and to ensure full ECCS operation immediately following loss of SCM. This is the highest priority operator action following such immediate responses to a reactor trip as ensuring control rods have been inserted. In addition, ECCS pumps may not be throttled or turned off unless specific conditions are met. These actions are basic to B&W plant emergency response and operators are repeatedly trained in the actions and the need for such actions. FENOC has proposed changing some of these emergency response instructions for this TS exception. The staff investigated the proposed actions to assess the potential for operator confusion due to the change of guidance philosophy, and the potential for core uncovery. The results of the staff's evaluation of FENOC's proposed changes are summarized in Sections 3.4.4 and 3.4.5, below.

3.4.4 Considerations Involving Throttling and Termination of RCS Injection

As discussed in Sections 3.1 - 3.3, above, there are substantial differences between the predicted DBNPS response to postulated LOCAs during power operation and during the proposed tests. The effect of these differences may be summarized as follows:

- Most LOCAs will be mitigated without a need to resort to HPI throttling. The justification for this conclusion was assessed in Sections 3.1 - 3.3, above. An approximate prediction of the likelihood of entering the applicable throttling procedures is provided in Section 3.4.6, below.

- In cases where HPI throttling may be needed, FENOC calculations predict the core will remain covered and pressure will fall until cooling is provided by LPI. This would not occur in corresponding LOCAs from power operation. Core coverage will ensure fuel cladding temperatures never significantly exceed core water temperature because the cladding heat fluxes are small. Core uncovery would occur if the corresponding LOCAs had occurred from power operation.

- The LOCA sizes of concern are sufficiently small that BWST depletion will take hours. MU pump throttling is anticipated to be started about two hours after LOCA initiation, with MU pump operation terminated over the next hour. HPI pump throttling, if needed, would be initiated following termination of MU pump operation. During this time, the Technical Support Center (TSC) would be activated to assist in diagnosis of system behavior and to provide additional operator guidance should such guidance be needed. This time would also be sufficient for the Nuclear Regulatory Commission's response center to be informed of the LOCA. LOCAs of similar size that occurred during power operation would progress more rapidly.

- Core coverage can be verified by incore temperature indications remaining at the saturation temperature. An indication of superheat would flag that core uncovery had occurred and operator remedial action was needed. Following power operation, significant core uncovery would occur before superheat was indicated because rapid boiling in the core would cause a high froth level with resultant wetting of the fuel cladding and incore temperature sensors. The boiling rates that are possible under the proposed Mode 3 test conditions are small (~2 gpm) and there is little difference between froth level and collapsed liquid level.

- Core heatup rate in the event of uncovery is slow. This will minimize the effect of thermal lag on incore temperature indication should uncovery occur during the proposed Mode 3 test. Greater than 5 hours are available following uncovery before cladding temperatures approach values that would begin to jeopardize cladding integrity. Heatup rates following power operation would be 50 to 100 times faster than during the proposed Mode 3 test, and temperature indication lag following power operation would be significant.

The last item is of particular interest because it allows time for operator action should the FENOC calculations contain a mistake or if the operators made a mistake and throttled HPI flow too aggressively. Operator action would be to increase HPI flow rate until the superheat condition was alleviated, and then to re-initiate throttling. MU pumps would provide a backup for HPI pumps if available, and if needed. The reduced sensible heat remaining in the RCS would be less during the recovery action, a condition that would contribute to successful pressure reduction.

The staff requested additional information regarding control room personnel resources for operations conducted during the proposed Mode 3 test, including operation in Mode 4. In response, the licensee, FENOC stated:

> The DBNPS TS require that for operation in Mode 4 and above, the minimum shift crew composition consists of two licensed Senior Reactor Operators (SROs), two licensed Reactor Operators (ROs), and the Shift Technical Advisor (STA) (who may be one of the two licensed SROs). The DBNPS TS also require that at least one licensed operator be in the control panel area when fuel is in the reactor, and at least two licensed operators, one with an SRO license, be present in the control room while in Modes 1 through 4. The DBNPS implementing procedures require at least one SRO and two ROs to be present in the control room during operation in Modes 1 through 4. During a typical startup from Mode 5, the control room is normally manned by the Shift Manager (an SRO), the Shift Engineer (an STA), the Unit Supervisor (an SRO), and two or more ROs, depending on what evolutions are in progress. Those licensed individuals beyond the procedural minimum requirements are permitted to leave the control room to address issues as needed. The commitment made in DBNPS letter Serial Number 2950 requires an additional licensed operator (beyond the TS required minimum of two SROs and two ROs) to be added to the shift complement. The additional licensed operator will be present in the control room during operations in Mode 4 and Mode 3 for the ... test.

3.4.5 Operator and TSC Personnel Training

FENOC proposed several procedural changes to address circumstances that may exist during the proposed Mode 3 test, including actions involved in taking the plant from Mode 5 to Mode 3 and returning it to Mode 5 following the Mode 3 test. These changes include the following:

(1) For some smaller break sizes, RCS depressurization to achieve an acceptable LPI flow rate and allow termination of HPI flow will be performed. There is a potential for this to involve HPI throttling without maintaining SCM.

(2) Opening the LPI cross-tie line may be necessary to provide flow to both CFT lines.

(3) Initiation of LPI flow through the APS line may be needed (a) to achieve long term boric acid concentration control or (b) if Item 2 cannot be accomplished following occurrence of a CFT line break.

FENOC's planned EOP modification is to add a step early in the EOP sections that could be entered during the proposed test. These steps will reference a new EOP attachment that will provide direction based on the guidance reviewed in this SE. The staff notes that adding one step in each of several EOP sections results in a minimal modification to the existing EOPs, and one that is easily reversed when it is no longer needed.

Review of the description of the new EOP attachment established general agreement with the actions described in References 1 and 2 that have been reviewed herein. There is one item, however, that is not clear and requires further examination. In several locations, the licensee discussion of the planned EOP attachment contains the following statement:

> Once the HPI shutdown criterion can be met, HPI is removed from service. The shutdown criterion is 500 gpm flow to each LPI line. If one train of LPI is not available, the HPI shutdown criterion can be met by placing the LPI cross-connect piping in service (By opening valve DH 830 or DH 831).

As shown in Section 3.4.2, above, smaller breaks, even with an open PORV and open vent valves, will not result in a 500 gpm LPI flow rate in each LPI line. The above criterion theoretically would not allow HPI to be removed from service. The staff notes, however, that this does not cause a significant problem for the following reasons:

(1) With respect to the licensing basis, the approach is to achieve LPI flow prior to initiating recirculation from the containment emergency sump. Once this is accomplished, as established herein, there is no further need for HPI and, assuming HPI is not increased to repressurize the RCS, considerations regarding HPI operation are irrelevant.

(2) With respect to actual operations, the EOPs provide guidance until the plant is placed in a condition where other procedures may be used. When LPI flow is achieved and RCS pressure is ensured to remain somewhat below the LPI pump shutoff head, plant management and TSC personnel will be able to provide long-term guidance to use procedures that reasonably ensure continued long-term cooling without HPI, and the HPI pumps can be removed from service.

FENOC committed to train potentially affected operators and TSC personnel on throttling and terminating MU and HPI prior to participating in operations associated with the Mode 3 test. Affected operators and TSC personnel will be retrained on normal operation of the MU and HPI pumps prior to participating in operations in Modes 1, 2, 3, or 4 after return of the plant to Mode 5 following completion of the proposed test.

3.4.6 Estimated Likelihood of Implementing HPI Throttling

FENOC has stated that testing and maintenance activities that would adversely impact vital system capabilities will be limited during the proposed Mode 3 test. This will:

- decrease the potential for initiation of LOCAs due to such activities as valve manipulations,

- increase the likelihood that normal makeup and useful support equipment would be available if needed, and

- tend to assure that other, more preferred, options for cooldown and depressurization would be available, thus reducing the likelihood of implementing HPI throttling to accomplish RCS depressurization.

In response to a staff question, FENOC provided a list of the presently anticipated testing and maintenance activities that could potentially affect equipment required for mitigation of LOCAs during the proposed Mode 3 test. The listed items are required by TS. FENOC further stated that "additions to the outage schedule are reviewed by the Scope Review Team prior to incorporation in the 13th Refueling Outage schedule. The Scope Review Team is chaired by the Shift Outage Director and consists of representatives from Outage Scheduling, Operations, Maintenance, Engineering, Radiation Protection, Materials, and Shutdown Risk. Work Activities are reviewed against two criteria; 1) the work is required to support primary or secondary plant startup, or 2) the work is required by Technical Specifications, Ongoing Commitment, Code, etc. Implementation of the May 14, 2003, commitment concerning maintenance activities and surveillance testing will ensure that any additional maintenance or testing will be appropriately evaluated prior to being conducted during the...test."

With consideration of the above background information, the staff made the following assessment to estimate the likelihood that operators would implement the HPI throttling process during the proposed Mode 3 test:

Probability of	Probability
LOCA during test	(2) / (52) $X10^{-2}$
LOCA will require throttling if primary depressurization actions fail	10^{-1}
RCPs, SG cooling, and ECCS throttling with SCM not unsuccessful	10^{-2}
Not depressurizing via PORV and high point vents with no SCM	10^{-2}
HPI throttling with loss of SCM (combination of above)	10^{-8}

The staff estimates that the likelihood of actually implementing the proposed HPI throttling procedures under loss of SCM conditions is small.

3.4.7 Assessment of HPI Pump Throttling

As established in Section 3.4.2, above, FENOC has predicted, using approved EM methods, that HPI throttling will result in a pressure decrease to achieve LPI cooling without significant core uncovery. HPI throttling may require operator actions that differ from those that are in place for response to LOCAs that initiate from power operation. Consequently, the staff has examined FENOC proposed actions to accomplish HPI throttling. The following is an assessment of the staff's examination:

- FENOC's planned EOP modification is to add a step early in the EOP sections that could be entered during the proposed test. These steps will reference a new EOP attachment that will provide direction based on the guidance reviewed in this SE. Adding one step in each of several EOP sections results in a minimal modification to the existing EOPs, and one that is easily reversed when it is no longer needed. This is an acceptable approach.

- In comparison to transient response immediately following power operation, plant transient response is slow during conditions where HPI throttling may be needed, personnel in excess of TS requirements will be provided, and TSC personnel will be available in time to provide additional throttling and related guidance.

- FENOC states that the planned operator actions can be taken from the control room and control room instrumentation provides sufficient information for the planned operator responses.

- Operators and TSC personnel will be trained for HPI throttling in the event it should be needed.

- FENOC has addressed potential concerns in regard to necessary testing and surveillance activities during the proposed Mode 3 test.

The staff finds that the above described actions provide reasonable assurance that HPI is not necessary during recirculation from the containment emergency sump in the event of LOCAs that could reasonably initiate during the proposed Mode 3 test. Consequently, HPI pump operability during recirculation from the containment emergency sump is not necessary to meet the requirements of 10 CFR 50.46 during the Mode 3 test.

Further, affected operators and TSC personnel will be retrained on normal operation of the MU and HPI prior to participating in operations in Modes 1, 2, 3, or 4 after return of the plant to Mode 5 following completion of the proposed test. This, and the approach that minimizes changes to existing EOPs so that the changes are easily removed, alleviates potential staff concerns with a change that is contradictory to basic mitigative actions contained in the existing EOPs. Finally, the likelihood of actually implementing the proposed HPI throttling procedures under loss of SCM conditions is small.

Consequently, the staff concludes that the proposed HPI throttling actions are acceptable.

3.5 HPI Pump Operability Concerns

3.5.1 Potential HPI Pump Deadheading

LER 2003-003 (Reference 2) identified a potential condition where RCS pressure may be higher than the discharge pressure from the HPI pump when operating in the piggyback configuration with water from the containment emergency sump. In this condition, there would be no flow of ECCS water through the HPI pump and, since ECCS water is used for HPI pump cooling, pump overheating could result. In Reference 2, FENOC stated that “a minimum flow recirculation flow path will be provided, if required, for the HPI pumps when operating in piggy-back alignment on containment emergency sump recirculation.” This commitment, when implemented, will address the potential pump overheating concern.

3.5.2 Potential for Debris When Configured To Take Suction from the BWST

In Reference 2, FENOC stated that “operation of the HPI pumps using the BWST as a suction source has not resulted in failures to date. Inspection of the HPI pumps during the 13th Refueling Outage (13RFO) did not show signs of wear attributable to debris.” Consequently, the staff concludes that unmodified HPI pump operability due to debris is not a significant concern when configured to take suction from the BWST.

3.6 Potential for Accidents that Differ from Power Operation Accidents

The staff assessed both licensing basis accidents and accidents that may be outside the envelope defined by licensing basis accidents to determine if there were any conditions associated with the proposed Mode 3 testing that could be more severe than encountered during power operation. The staff’s findings are summarized in the following subsections to Section 3.6.

3.6.1 IMI Nozzle Breaks

The licensee stated that, in conjunction with Framatome ANP, FENOC concluded that IMI nozzle leakage is highly unlikely. The licensee also stated that it considered the normal operating pressure leakage test to be confirmatory in nature.

A partial pressurization and inspection of the RCS in accordance with the Restart Test Plan will be performed to confirm that there are no visible leaks prior to entering Mode 3. This partial pressurization and inspection process is a commitment that contributes to a reduction in the likelihood of IMI failure during the Mode 3 test.

The licensee addressed the effect of IMI nozzle breaks by stating that “the break area of a single IMI nozzle, based on the inside diameter (ID), is 0.0021 ft^2. (The effective break area based on the outside diameter (OD) is estimated to be approximately three times the area based on the inside diameter.) The DBNPS can mitigate the consequences of a 0.0085 ft^2 break from a core power of 1.02 times 2966 MWt (power uprate level) with typical LOCA assumptions and the most limiting single failure. This includes no credit for operator initiated secondary side depressurization. Under Mode 3 conditions, the break size that could be mitigated is similar to that for the size that can be mitigated under full power conditions. Thus, the equivalent of one ID and one OD limited break, or four ID limited nozzle breaks have been shown to provide acceptable results.”

The staff finds that FENOC has acceptably addressed potential IMI breaks with respect to the conduct of the proposed Mode 3 test and the amendment request.

3.6.2 Other Potential Accidents

A generic issue is being addressed by the staff and the B&W Owners Group where certain postulated LOCAs in an upper hot leg may lead to a rapid cooldown of the SG tubes while the SG shell remains hot. The resulting temperature difference may overstress the SG tubes. The low decay heat generation rate that will exist during the proposed Mode 3 test would result in the ECCS cooling the tubes about 20 °F to 40 °F more at the time of maximum tube stress than would occur during power operation. This effect is offset by a SG shell temperature that is about 25 °F less than is the case for power operation. The probability of this condition is extremely small, and the staff believes the likelihood of tube breakage is also small. The staff concludes that there is no significant difference between the proposed test and power operation. Therefore, the staff finds this generic issue is not of concern during the proposed Mode 3 test.

3.7 Makeup Pump Operation

The licensee stated that “two MU pumps are required to be operable in Mode 3 and in Mode 4 with RCS pressure above 150 psig by TS 3.1.2.4. Both MU pumps and both HPI pumps will be available (i.e., capable of being placed in service within the time they are required)” while operating with suction from the BWST. The MU pumps are not intended to circulate fluid from the containment emergency sump.

The MU pumps are powered from the same buses that supply power to the HPI and LPI pumps. Hence, they can be operated from either offsite or onsite power supplies. However, the MU pump room ventilation is not powered from an essential electrical bus, which limits how long the pumps could be operated with a loss of offsite power without compensatory measures.

Operators are instructed to maximize MU flow by running both MU pumps if pressurizer level has decreased to less than 40 inches or SCM is lost. Thus, MU pumps will be used as part of the equipment for mitigating a LOCA while taking suction from the BWST or from the MU tank. The normally controlled MU tank level will provide about 1150 gallons to 2100 gallons of borated water for injection into the RCS. MU tank inventory can be replenished from the boric acid addition tanks (BAATs), the demineralized water storage tank, or the clean liquid radioactive waste receiving tanks (CWRT) (Clean Waste Receiving Tank 1 or 2, or Clean Waste Monitor Tank 1 or 2). Makeup capability of some of these sources to the BWST is covered in Section 3.9, below.

If throttling is necessary to reduce RCS pressure to obtain LPI before transfer to recirculation from the containment emergency sump, then MU pump throttling is anticipated to be started about 2 hours after LOCA initiation, with MU pump operation terminated over the next hour. HPI pump throttling, if needed, would be initiated following termination of MU pump operation.

Note restarting MU pumps is a defense-in-depth measure if later injection is needed and HPI pumps have been lost. Further, MU pump shutoff head is in excess of the pressurizer safety valve settings so that successful MU pump operation ensures water can reach the RCS unless the MU piping is broken.

3.8 Mode 4 Operations

The DBNPS TS contain no ECCS requirements for Mode 4 operation. After discussion with the staff, FENOC made the following commitments (Reference 2):

- Both MU pumps and both HPI pumps will be available during Mode 4 operation.

- Both HPI pumps will be available for use in a piggy-back configuration if a decision is made to operate them in that configuration.

- Any loss of equipment required for operation in Mode 3 that results in a transition to Mode 4 will be restored or the plant will be placed in Cold Shutdown (Mode 5) within 24 hours following entry into Mode 4.

The first two commitments satisfied staff concerns regarding ECCS capability for test-associated operation in Mode 4. The last commitment satisfactorily addressed staff concerns that an equipment failure could lead to long-term operation in Mode 4 where ECCS capability might be diminished.

3.9 BWST Makeup Capability

The DBNPS currently has procedures in place that describe the ability to add borated water to the BWST from the BAATs, CWRTs, Clean Liquid Radioactive Waste Monitor Tanks (CWMTs), and Spent Fuel Pool (SFP). The approximate flow rates and estimated times to place in service are as follows:

Source	Flow Rate	Time
BAAT	40 gpm	~30 minutes
CWRT	120 gpm	~1 hour
CWMT	120 gpm	~1 hour
SFP	100 gpm	~1 hour

Additions from the BAAT also require additions from demineralized water due to the high boron concentration of the BAAT. Additions from these sources require local valve manipulations in the Auxiliary Building. Volumes available from each source will vary, depending on the DBNPS operational activities.

3.10 Boron Precipitation Control (BPC) during Long Term Cooling

The EOPs instruct the operators to initiate active boron concentration control methods after switchover to emergency sump recirculation and: (1) adequate SCM does not exist, (2) average incore thermocouple temperature is less than 333 °F, and (3) RCS pressure is less than or equal to 200 psig.

In the existing licensing basis, the primary method for BPC uses an HPI pump to supply water to the APS line. The backup method uses an operating LPI pump taking suction from the DHR drop line with discharge into the RV via the core flood nozzles. The primary method would not be operable under the proposed exception. The backup method would be unaffected. As a substitute for the primary method, DBNPS proposes aligning an LPI pump to feed the APS line. The APS flow rates vary depending on LPI flow to the reactor vessel, but preliminary FENOC estimates show that at RCS pressures less than 92 psia, the APS flow rates should be slightly more than 30 gpm at the highest pressures up to approximately 70 gpm at atmospheric pressure. Assuming an APS flow rate of 40 gpm, it would take approximately 5 hours to fill the pressurizer. Flow into the hot leg and the RV would begin after the pressurizer was filled.

The LPI train cross-connect valves are normally operated from the control room. In the event of a loss of an emergency diesel generator (EDG) and assuming offsite power has been lost, the LPI train cross-connect is still capable of being placed in service from the control room.

Establishing LPI flow through the APS line requires manual valve manipulations by the operator. The valves that must be operated are located in the same mechanical penetration room. Although a verification walkdown has not been performed, FENOC expects that the valves could be repositioned within 30 minutes of the decision to initiate LPI flow through the APS line. In the event of a loss of an EDG and assuming offsite power has been lost, additional manual operations to provide power to the affected motor-operated valves in the APS flow path would be required. Walkdowns have shown that these additional operator actions can be completed within approximately 23 minutes. FENOC expects the areas in which the manual actions to initiate LPI flow through the APS line are performed to be accessible following a LOCA as the FENOC evaluation shows that no fuel damage is expected to occur.

Considering the low decay heat generation rate, the slow heatup rates, and the low boiling rate assuming adiabatic conditions, it is clear that a few gpm of core flow (>10.5 gpm using licensing basis assumptions) will prevent significant boric acid precipitation. The staff concludes that boric acid control during long-term cooling is acceptable for this request.

4.0 COMMITMENTS

The licenses May 14 and August 2, 2003 letters provided the following commitments:

The commitments are as follows:

Commitment	Implementation
(1) If the HPI pumps are modified to incorporate internal strainers, a determination will be made that the HPI strainers will not clog while injecting water from the BWST to the RCS as follows:	Prior to operation in Mode 4 and Mode 3 under the exception proposed by license amendment request (LAR) 03-0008.
(a) The BWST will be recirculated using the borated water recirculation pump at a flow rate of approximately 170 gpm for a time sufficient to ensure that two BWST volumes have been recirculated. A representative sample will be taken, and filtered through filtration media of sufficient size to ensure that debris that may clog the modified HPI pump strainers is collected. The sample will be inspected to determine the type of debris collected.	
(b) The BWST will then be recirculated using an LPI pump at a flow rate of approximately 3000 gpm for a time sufficient to ensure that two BWST volumes have been recirculated. Representative samples will be taken at the LPI pump suction, and filtered through filtration media of sufficient size to ensure that debris that may clog the modified HPI pump strainers is collected. The sample will be inspected to determine the type of debris collected.	
(c) With the associated LPI pump recirculating the RCS, each HPI pump will be operated in the piggy-back mode of operation at a flow rate of approximately 950 gpm. Representative samples will be taken at the HPI pump suction, and filtered through filtration media of sufficient size to ensure that debris that may clog the modified HPI pump strainers is collected. The sample will be inspected to determine the type of debris collected.	
(d) Each HPI pump will be aligned to recirculate the BWST at a flow rate of approximately 430 gpm. Representative samples will be taken at the HPI pump suction, and filtered through filtration media of sufficient size to ensure that debris that may clog the modified HPI pump strainers is collected. The sample will be inspected to determine the type of debris collected.	
(e) The section of piping between the containment emergency sump and valves DH9A and DH9B will be inspected for debris.	
(2) Both MU pumps and both HPI pumps will be available (i.e., capable of being placed in service within the time they are required). In the case of HPI pump operation in piggy-back alignment with a LPI pump, operation in the presence of sump debris is not assured.	During operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008.
(3) A minimum flow recirculation flow path will be provided, if necessary for HPI pump cooling, for the HPI pumps when operating in piggy-back alignment on containment emergency sump recirculation.	

Commitment	Implementation
(4) Potentially affected operators will be trained on the throttling and termination of MU and HPI prior to participating in operations associated with the Mode 3 test.	As stated
(5) Affected operators will be retrained on normal operation of MU and HPI prior to participating in operations in Modes 1, 2, 3, and 4 after return of the plant to Mode 5 following completion of the proposed Mode 3 test.	As stated
(6) Potentially affected TSC staff will be trained on the basis for throttling and termination of MU and HPI prior to participating in operations associated with the Mode 3 test.	As stated
(7) Affected TSC staff will be retrained on normal operation of MU and HPI prior to participating in operations in Modes 1, 2, 3, and 4 after return of the plant to Mode 5 following completion of the proposed Mode 3 test.	As stated
(9) During operation under this limited exception, for any loss of equipment required for operation in Mode 3 that results in a transition to Mode 4, the affected component will be restored or the plant will be placed in Cold Shutdown within 24 hours following entry into Mode 4.	During operation in Mode 3 under the exception proposed by LAR 03-0008.
(10) An additional licensed operator (beyond the TS required minimum of two SROs and two ROs) will be added to the shift complement and will be present in the control room during operations in Mode 3 and Mode 4 for the proposed test. (See the end of Section 3.4.4, above.)	As stated
(11) A partial pressurization and inspection of the RCS in accordance with the Restart Test Plan will be conducted prior to entering Mode 3. (See Section 3.6.1, above.)	As stated
(12) “If IMI nozzle leakage is discovered, the proposed exception would not be utilized for a Mode 3 entry following corrective actions.” (Reference 1)	As stated
(13) “Operation in Mode 1 or Mode 2 while relying upon the provisions of this exception is prohibited.” (Reference 1)	As stated
(14) Existing EOPs will be modified to cover potential depressurization operations applicable to the proposed Mode 3 test and a new EOP section will be written that is referenced in the EOP modifications. These EOPs will be verified to be correct and will be in the control room and the TSC prior to entering Modes 4 or 3 under the proposed exception. (See Section 3.4.7, above.)	As stated

Commitment	Implementation
(15) Maintenance activities and surveillance testing that could affect response to a LOCA during conditions where the proposed exception applies will be minimized and conducted in accordance with the FENOC Reference 2 description. (See Section 3.4.6, above.)	As stated

5.0 SUMMARY

The staff has reviewed FENOC's request for an amendment to add an exception to the DBNPS TS for LCO 3.5.2 to remove the requirement for operable HPI pumps during its proposed Mode 3 test. The staff's findings as a result of this review are as follows:

(1) As discussed in Section 4, above, there are 15 commitments that have been made by FENOC. These commitments are entered into the licensee's Regulatory Commitment Tracking System. This is acceptable to the NRC staff.

(2) The licensing basis decay heat generation rate required by 10 CFR Part 50 Appendix K is a factor of five greater than the realistic heat generation rate. FENOC's licensing basis value for the decay heat fraction of full power of 0.00089 is conservative and is acceptable.

(3) The small inventory of radioactive material means that the existing licensing basis release predictions for accidents that are postulated to initiate during full power operation far exceed releases that would be predicted for DBNPS for similar accidents. Further, no other accidents were found where core damage could exceed what is predicted for design-basis accidents that initiate during full power operation. Consequently, no further consideration of release and dose is necessary because the existing licensing basis bounds the releases that can be postulated as originating during the proposed Mode 3 test.

(4) The FENOC predictions of RCS pressure response to a LOCA are based on typical licensing basis assumptions of one ECCS train with degraded ECCS pump characteristic curves. The calculated RCS pressure as a function of time will typically provide a lower bound of the pressure behavior. Behavior for an increased injection flow rate may be obtained by assuming the break flow area is increased in direct proportion to the increase in injection flow rate. Consequently, the FENOC calculations may be applied to a range of RCS injection capabilities and the calculations are acceptable for predicting pressure for conditions that may be postulated to exist during the proposed Mode 3 test.

(5) The licensing basis LOCAs of potential concern during the proposed Mode 3 test will have a high enough flow rate to remove all decay heat through the break without causing an RCS heatup or RCS repressurization. Most LOCAs will be mitigated without a need to resort to HPI throttling.

(6) In cases where HPI throttling may be needed, FENOC calculations predict the core will remain covered and pressure will fall until cooling is provided by LPI. Core coverage will ensure fuel cladding temperatures never significantly exceed core water temperature because the cladding heat fluxes are small. If the operators were to make a mistake and throttle too rigorously, incore temperature indicators would flag that core uncovery had occurred and operator remedial action was needed. More than 5 hours are available

following uncovery before cladding temperatures approach values that would begin to jeopardize cladding integrity. The slow heatup times would provide adequate time for operator action.

(7) The LOCA sizes of concern where throttling may be necessary to ensure reaching LPI prior to initiating recirculation from the containment emergency sump are sufficiently small that BWST depletion will take hours. MU pump throttling is anticipated to be started about 2 hours after LOCA initiation, with MU pump operation terminated over the next hour. HPI pump throttling, if needed, would be initiated following termination of MU pump operation. During this time, the TSC will have been fully activated to assist in diagnosis of system behavior and to provide additional operator guidance should such guidance be needed.

(8) Consistent with the Item (4) finding, FENOC stated that "consideration of the maximum ECCS flows, including two MU pumps, was included in the throttling guidance.... Therefore, the report covers both minimum and maximum ECCS flow without exacerbating the severity of the problem or under-predicting the boundary conditions that support a successful demonstration of the 10 CFR 50.46 criteria." The planned throttling guidance acceptably addresses the anticipated RCS pressure response following initiation of a LOCA.

(9) Heat loss from the RCS pressure boundary, in the complete absence of heat removal by flowing water, will result in a maximum RCS temperature that is less than normal power operating temperature. If an adiabatic system is assumed, the boiling rate that is possible under the proposed Mode 3 test conditions is approximately 2 gpm. For comparison, typical power operating conditions would result in a boiloff rate of over 100 gpm. These examples show that postulated LOCAs initiating from the Mode 3 test conditions will be less challenging than those initiating from power operation.

(10) FENOC's planned EOP modification is to add a step early in the EOP sections that could be entered during the proposed test. These steps will reference a new EOP attachment that will provide direction based on the guidance reviewed in this SE. Adding one step in each of several EOP sections results in a minimal modification to the existing EOPs, and one that is easily reversed when it is no longer needed.

(11) FENOC has acceptably addressed potential IMI breaks with respect to the conduct of the proposed Mode 3 test and the amendment request.

(12) The hot leg LOCA followed by tube rupture generic issue is not of concern during the proposed Mode 3 test.

(13) FENOC has satisfied licensing requirements for consideration of boric acid control during long-term cooling.

Consequently, with respect to regulatory requirements:

(1) The findings summarized above provide reasonable assurance that HPI is not necessary during recirculation from the containment emergency sump in the event of postulated LOCAs that initiate during the proposed Mode 3 test. Therefore, HPI pump operability during recirculation from the containment emergency sump is not necessary to meet the requirements of 10 CFR 50.46 during the Mode 3 test. Further, the GDC 35 requirement to provide abundant emergency core cooling, consistent with meeting the single-failure

criterion, is satisfied.

(2) The small inventory of radioactive material means that the existing licensing basis release predictions for accidents that are postulated to initiate during full power operation far exceed releases that would be predicted for DBNPS for similar accidents. Therefore, the staff finds that 10 CFR Part 100 is satisfied.

(3) The findings establish that 10 CFR 50.57(a)(3) continues to be met.

Based on the considerations discussed above, the staff concludes that the proposed amendment is acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 24668). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. Myers, Lew W., "LER 2003-002, Davis-Besse Nuclear Power Station, Unit No. 1, Date of Occurrence - October 22, 2002," Letter to NRC from FirstEnergy transmitting "Potential Degradation of High Pressure Injection Pumps Due to Debris in Emergency Sump Fluid Post Accident," Licensee Event Report 2003-002-00, Davis Besse Unit Number 1, NP-33-03-002-00, May 5, 2003.

2. Bezilla, Mark B., "LER 2003-003, Davis-Besse Nuclear Power Station, Unit No. 1, Date of Occurrence - September 25, 2002," Letter to NRC from FirstEnergy transmitting "Potential Inadequate HPI Pump Minimum Recirculation Flow Following SBLOCA," Licensee Event Report 2003-003-00, Davis Besse Unit Number 1, NP-33-03-003-00, June 10, 2003.

3. Bezilla, Mark B., "Davis-Besse Nuclear Power Station Incore Monitoring Instrumentation Nozzle Inspections," Letter to NRC from FirstEnergy, Serial Number 2973, July 30, 2003.

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Date: September 5, 2003