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License Number NPF-3

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Subject: Davis-Besse Nuclear Power Station, Unit Number 1
Supplemental Information Regarding License Amendment Application to Revise Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems – ECCS Subsystems – $T_{avg} \geq 280°F$
(License Amendment Request No. 03-0008; TAC No. MB8953)

Ladies and Gentlemen:

On May 14, 2003, the FirstEnergy Nuclear Operating Company (FENOC) submitted an application for an amendment to the Davis-Besse Nuclear Power Station, Unit Number 1 (DBNPS) Operating License NPF-3, Appendix A Technical Specifications, regarding Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems – ECCS Subsystems – $T_{avg} \geq 280°F$. The proposed amendment (DBNPS letter Serial Number 2950) would add an exception applicable only during the Mode 3 (Hot Standby) Restart Test Plan normal operating pressure inspection activities conducted during the ongoing thirteenth refueling outage (13RFO).

The NRC provided a Request for Additional Information (RAI) to FENOC related to the NRC technical staff review of LAR 03-0008 on July 31, 2003. Completed responses, previously provided to the NRC staff on a preliminary basis, are provided in Enclosure 1 of this submittal. Additional information responding to the rest of the NRC staff questions is provided in Enclosure 2 of this submittal.

The information provided in Enclosures 1 and 2 of this submittal does not affect the conclusions stated in the previously submitted license amendment application that there is no adverse impact on nuclear safety and that the proposed amendment involves no significant hazards consideration.

With the providing of this additional information, FENOC requests that the NRC staff complete its review of this license amendment application and issue the license amendment by August 12, 2003. Should you have any questions or require additional information, please contact Mr. Kevin L. Ostrowski, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

CWS

Enclosures

cc: Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
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SUPPLEMENTAL INFORMATION IN SUPPORT OF

APPLICATION FOR AMENDMENT

TO FACILITY OPERATING LICENSE NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This submittal provides supplemental information with respect to the Davis-Besse Nuclear Power Station Unit Number 1, Facility Operating License Number NPF-3, License Amendment Request Number 03-0008 (DBNPS letter Serial Number 2950) dated May 14, 2003. The statements contained in this submittal, including its associated enclosures, are true and correct to the best of my knowledge and belief.

I declare under penalty of perjury that I am authorized by the FirstEnergy Nuclear Operating Company to make this request and the foregoing is true and correct.

Executed on: 8/2/03

By: Lew W. Myers
Lew W. Myers, Chief Operating Officer

Docket Number 50-346
License Number NPF-3
Serial Number 2980
Enclosure 1

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NUMBER 1 (DBNPS) LICENSE AMENDMENT APPLICATION TO REVISE TECHNICAL SPECIFICATION (TS) 3/4.5.2, EMERGENCY CORE COOLING SYSTEMS – ECCS SUBSYSTEMS – $T_{avg} \geq 280°F$ (LICENSE AMENDMENT REQUEST NO. 03-0008)

COMPLETION OF RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION

(10 pages follow)

COMPLETION OF RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION

2. The FENOC request identifies potential concern with leakage at the reactor vessel Incore Monitoring Instrumentation (IMI) nozzles. It does not address the other end of the incore monitoring tubes where the instrumentation transits from the Reactor Coolant System (RCS) pressure to containment pressure. Are these seal regions to be evaluated for potential leaks?

   DBNPS Response

   The Restart Test Plan referenced in the FirstEnergy Nuclear Operating Company (FENOC) May 14, 2003, submittal (Davis-Besse Nuclear Power Station (DBNPS) Letter Serial Number 2950) requires testing of the RCS, including components within the reactor coolant pressure boundary (RCPB) and associated piping exposed to full RCS pressure, to ensure integrity following replacement of the reactor vessel pressure head and maintenance of RCS piping and components. DBNPS procedure DB-PF-03010, RCS Leakage Test, is used for inspections of the RCPB and requires inspection of the portion of the RCS pressure boundary identified in the NRC staff technical reviewer's question.

3. FENOC is generally silent regarding Mode 4 operation where pressurization beyond the Low Pressure Injection (LPI) pump injection capability is possible. What RCS water makeup capability will be provided during Mode 4 and how will this capability be reasonably assured?

   DBNPS Response

   The DBNPS Technical Specifications (TS) do not require High Pressure Injection (HPI) pump operability in Mode 4. The proposed change to the TS affects the requirements in Mode 3 only. Therefore, there is no impact on the current Mode 4 requirements as a result of the proposed license amendment.

   The DBNPS safe shutdown state defined by the current licensing basis is Mode 3. DBNPS relies on non-safety grade components to transition between Mode 3 and Mode 5. Sources of RCS makeup (various combinations of makeup (MU) pumps and HPI pumps) are maintained available in order to reduce risks associated with shutdown operation in accordance with FENOC procedure NOP-OP-1005, Shutdown Safety. One MU pump operates to provide reactor coolant pump (RCP) seal injection and RCS makeup and would be available to inject water into the RCS. This MU pump is placed in service prior to entry into Mode 4. The second MU pump will also be available to supply RCS makeup. 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). The MU pumps are not intended to circulate fluid from the containment emergency sump (See the response to Question 6c). 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.

4. Will the HPI pumps be available for use in a piggy-back configuration if a decision is made to operate them in that configuration?

   DBNPS Response

   The HPI pumps will be available (i.e., capable of being placed in service within the time they are required) for use in a piggy-back configuration when ECCS is aligned for containment emergency sump recirculation as a defense-in-depth measure. 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.

6. With respect to makeup (MU) pumps:

   a. Can MU pumps be operated with power from the emergency diesel generators?

      DBNPS Response

      The MU pumps are both powered from redundant essential buses as described in DBNPS Updated Safety Analysis Report (USAR) Section 9.3.4.2.1, and thus are supplied from the emergency diesel generators. These are the same essential buses used to power the HPI and LPI pumps. However, the MU pump room ventilation is not essentially powered, limiting the duration of time that the pumps could operate under loss of offsite power conditions without other compensatory measures.

   b. Please address MU pump operation if a LOCA or other accident were to occur that would typically require HPI injection.

      DBNPS Response

      The operator is instructed to maximize MU flow by starting both MU pumps, if not already operating, if pressurizer level has decreased to less than 40 inches, or subcooling margin is lost. The operating MU pump(s) are automatically stopped upon detection of a bus undervoltage on the associated essential 4160 V bus with a Safety Features Actuation System (SFAS) Level 3 actuation signal (on high containment pressure of 18.7 psia, or low-low RCS pressure of 470 psig) present unless the associated LPI pump was manually started; or manually by the operator if an SFAS Reactor Coolant pressure < 450 psig Channel Trip (5-1-D) is received. If subcooling margin is not adequate and LPI flow to the RCS does not exist, the operator is directed to open the LPI piggy-back valves, placing the MU pumps in the piggy-back mode. For those LOCAs that do not result in receipt of an SFAS Level 3 actuation signal, or a Reactor Coolant pressure < 450 psig Channel Trip, the operator is directed to continue MU pump flow until just prior to

aligning the LPI suction to the containment emergency sump. At this time, the MU pump suction is transferred to the MU tank and the MU pumps are stopped.

As discussed on page 36 of the Framatome-ANP (FANP) report attached to DBNPS letter Serial Number 2950, dated May 14, 2003, Emergency Core Cooling System (ECCS) throttling should begin with the MU pumps if they are in operation at 2 hours after break initiation. Over the next hour the MU pump flow should be terminated. If the RCS pressure is still elevated, the guidance continues with instructions for throttling the HPI pumps.

c. FENOC indicates MU pumps cannot be used when BWST inventory has been depleted. Is it not possible or practical to transfer water from the containment emergency sump to the BWST where it would be available to the makeup pumps? (4.2, page 12)

DBNPS Response

The MU pumps are not intended to circulate fluid from the containment emergency sump. By letter dated October 27, 1987, (DBNPS letter Serial Number 1433) the DBNPS submitted a proposed license amendment request to support enhanced feed and bleed modifications. This submittal stated that the MU system is not intended to circulate fluid from the containment emergency sump; pump suction is restricted to either the MU tank or the BWST. The DBNPS committed to ensuring that pump suction is restricted to the MU tank and the BWST by modifying the appropriate procedures. The MU tank level is normally controlled between 55 inches and 85 inches. This provides approximately 1150 gallons to approximately 2100 gallons of borated water available for injection into the RCS. Transferring water from the containment emergency sump to the BWST would be expected to result in transferring debris to the BWST. Refilling the BWST is addressed in the response to Question 20. The MU tank can be filled from the Boric Acid Addition Tanks (BAATs), the Demineralized Water Storage Tank, or the Clean Radioactive Waste System (Clean Waste Receiving Tank 1 or 2 or Clean Waste Monitor Tank 1 or 2).

d. Do MU pumps trip upon ECCS initiation? The Framatome ANP Inc. 51-5026803-00 (referred to hereinafter as FANP) discussion appears to be mixed. Much of the discussion appears to exclude MU pump operation, yet Page 36 and other locations indicate they are running. If they are running, please discuss the RCS pressure response predictions since the predictions appear to be based on no MU and one HPI in operation.

DBNPS Response

The operating MU pump(s) are automatically stopped upon detection of a bus undervoltage on the associated essential 4160 V bus with a Safety Features Actuation System (SFAS) Level 3 actuation signal (on high containment pressure of 18.7 psia, or low-low RCS pressure of 470 psig) present unless the associated LPI pump was manually

started; or manually by the operator if an SFAS Reactor Coolant pressure < 450 psig Channel Trip (5-1-D) is received.

The discussions in Section 4 of the FANP report are based on the minimum ECCS analysis flows for a single HPI and LPI pump. Accordingly, Figures A-2 and A-3 of the FANP report give the pressure equilibrium conditions with one train of ECCS in operation. If two trains of ECCS are in operation, the break area will be double the area given in Figures A-2 and A-3. It would also be increased by any MU flow that may actually be available. When the total injection flow rate is increased, the RCS will equilibrate at a higher pressure. Consideration of the maximum ECCS flows, including two MU pumps, was included in the throttling guidance in Section 5.2 of the FANP report. 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.

In some places the FANP report is discussing the design basis response and in other places the report is discussing procedural implementation.

8. Near the end of Section 2.0, FENOC states that multiple Mode 3 entries may be necessary but then makes an exception following any corrective action if the IMI nozzle leakage is discovered. Please discuss the reasoning that underlies these potential actions.

DBNPS Response

As discussed in FENOC letter Serial Number 2973, dated July 30, 2003, FENOC does not believe that Incore Monitoring Instrumentation (IMI) nozzles leaks exist, therefore, the normal operating pressure (NOP) leakage test is confirmatory in nature. The intent of the statement is to allow multiple entries in the event other RCS leakage is detected or equipment conditions occur that require depressurizing to repair. Should leakage be found through the IMI nozzles, DBNPS will not enter Mode 3 again until after repairs to the IMI nozzle(s) have been made, issues with sump debris have been resolved, and the HPI pumps have been declared operable such that the exception does not apply.

9. Natural circulation is discussed in Section 4.1.1, but boiler-condenser operation is not mentioned. Omission of boiler-condenser operation also appears to apply to the FANP discussion. Please address the influence of boiler-condenser operation.

DBNPS Response

Boiler condenser mode (BCM) of operation is a key phenomenon in certain small-break loss-of-coolant accidents (SBLOCA) analyses. BCM heat removal may be required whenever the break energy discharge cannot meet or exceed the net system energy addition rates (core decay heat generation, reactor coolant pump energy addition, ECCS energy addition, etc).

For this low power Mode 3 NOP test, the decay heat can be removed with any break size that could be characterized as a LOCA.

In order for BCM to occur, the RCS level inside the steam generator (SG) tubes must be at or below the auxiliary feedwater (AFW) injection elevation, and the AFW must be flowing (high elevation BCM). Alternatively, if AFW is not flowing, the RCS level must be below the secondary side pool level (pool BCM). For BCM to occur, the primary side pressure must be above the secondary side pressure. When the core power is significant, BCM is critical for controlling RCS pressure during a SBLOCA. However, when there is virtually no core power, there is little potential for anything other than a very brief period of BCM heat removal. BCM cannot be continuously supported because there is insufficient steam generated on the primary side.

10. Do conditions exist that are unique to the FENOC request that would exacerbate the hot leg LOCA issue discussed in BAW-2374?

    DBNPS Response

    The loss of credited HPI flow during the sump recirculation phase does not exacerbate the hot leg LOCA issue. The lower decay heat associated with the long period of shutdown prior to this Mode 3 NOP test, however, would allow the ECCS to cool the SG tubes more quickly compared to typical post-trip core power levels. The estimated decrease in the tube average temperature is on the order of 20 to 40°F. The SG shell average temperature, however, would also be cooler during this Mode 3 NOP test by approximately 25°F over the full power value. This lower shell temperature would reduce, or somewhat compensate for, the lower tube temperature. As a result, the shell-to-tube temperature difference would be approximately 15°F higher than the 374°F calculated at the full power conditions. This variation is a generic issue for any plant with lower core power levels and is not unique to the FENOC request.

11. We need additional information in regard to RCS heat loss. The following apply:

    d. Please address the trade-off of running reactor coolant pumps (RCPs) versus tripped RCPs and RCS depressurization with consideration given to RCS heat rate loss.

       DBNPS Response

       The RCPs will remain in operation unless offsite power is lost or the operators trip the pumps. Typical LOCA applications consider the loss of offsite power at the time of turbine trip following the LOCA because of the perturbation of the grid when the plant trips. For this Mode 3 scenario, the plant is not producing power such that a reactor trip would cause the grid to be perturbed. In this case, the operators will let the RCPs run unless subcooling margin is lost. When the RCPs are in operation, the steam generators can be used to cool and depressurize the entire RCS. If subcooling margin is lost, the

pumps must be tripped within 2 minutes to prevent continuous RCS liquid discharge out of the break. Consequently, RCP operation while the RCS is saturated is not permitted and is not endorsed for this NOP test.

e. We do not understand the FANP Page 28 discussion that "there is insufficient decay heat to create substantial natural circulation flow rates that can cool the bulk of the RCS fluid." Please explain.

DBNPS Response

Robust natural circulation in the B&W-designed plant is created by thermal gradients produced by the core heat generation and steam generator heat removal. With extremely low decay heat levels, the break energy discharge will remove all the core heat generation, and steam generator heat removal is not needed. Therefore, significant natural circulation flows will not be obtained. The heat addition rates that do occur will result in low natural convection flows that may not be sufficient to push cold water out of the cold leg pump suction (CLPS) piping regions. Therefore, there may be pockets of hot and cold liquid in the reactor vessel and RCS piping that are not easily cooled uniformly.

12. On a realistic basis, please address availability of turbine-driven auxiliary feedwater pumps from the viewpoint of available RCS thermal energy.

DBNPS Response

The steam generators are the typical heat sink for the core. In this case, the core power is so low that RCS heat losses are sufficient to remove the core power. As a result, there is little need for steam generator heat removal following a SBLOCA, and therefore the availability of the turbine-driven auxiliary feedwater pumps, which feed the SGs, does not factor into a realistic evaluation of RCS cooling. In addition, see the response to Question 11.a in Enclosure 2 to this letter.

20. What is the practical long-term borated water makeup capability to the BWST?

DBNPS Response

The DBNPS currently has procedures in place that describe the ability to add borated water to the BWST from the Boric Acid Addition Tank (BAAT), Clean Waste Receiving Tanks (CWRTs), Clean Waste Monitor Tanks (CWMTs), and Spent Fuel Pool (SFP). Approximate flow rates, and times to place in service are as provided below.

Source	Flow Rate*	Time*
BAAT	40 gpm	~30 minutes
CWRT	120 gpm	~1 hour

CWMT	120 gpm	~1 hour
SFP	100 gpm	~1 hour

*Flow rates and times are estimates per DBNPS Operations and have not been verified by walkdown.

Additions from the BAAT also require additions from demineralized water due to the high boron concentration of 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.

22. If the only available steam generator has a steam generator tube rupture, why can't it be used for cooldown? (4.2, Page 13)

DBNPS Response

Non-reliance on the ruptured SG was described in the discussions of the risk analysis. Relying on the ruptured steam generator could result in a release to the environment. In addition, this would result in "negative training" for the operator.

23. Item 2 of the No Significant Hazards Consideration asks "Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?" The FENOC response addresses accident initiators, but does not address differences in accident progress. Please clarify. (5.1, Page 18)

DBNPS Response

Normally, postulated accidents are analyzed to occur from full power conditions with bounding decay heat. In the Mode 3 evaluation performed for this license amendment request, although different operator actions are required to be credited these actions are similar to those assumed in the present DBNPS USAR for balancing HPI flow. The evaluation showed that either no clad temperature excursion is predicted to occur due to no core uncovery, or the excursion is bounded by the Mode 1 cases previously analyzed.

26. The proposed footnote to TS Section 3.5.2 states "Under this exception, neither HPI train is required to be capable of taking suction from the LPI trains when aligned for containment sump recirculation." This appears to imply that the HPI trains may not be aligned for piggy-back operation and may not be capable of such an alignment. Please explain why FENOC does not provide wording consistent with requiring that HPI will be capable of being aligned and operated in the piggy-back configuration while recognizing that operability is not ensured.

    DBNPS Response

    An exception to the operability requirements, as provided in the proposed license amendment, is considered to be less confusing to the TS user than a discussion requiring the piggy-back capability while recognizing that operability is not ensured. However, as stated in response to NRC Question 4, the HPI pumps will be available for use in a piggy-back configuration as a defense-in-depth measure.

27. Note 1 of Table 3-1 of FANP states that SG heat removal can be accomplished with non-safety grade equipment. Does this mean SG heat removal cannot be accomplished with only safety grade equipment?

    DBNPS Response

    There is safety-related SG heat removal via AFW and main steam safety valves. However, SG secondary side depressurization (other than with the turbine-driven AFW pump steam flow) is accomplished with non-safety grade equipment. The atmospheric vent valve system is not considered to be safety-grade, although the valves themselves are safety-grade. As indicated in the response to Question 42 in Enclosure 2, the Emergency Operating Procedure will direct the use of the atmospheric vent valves, as appropriate.

28. We believe FANP Section 3.2.2 states that the normalized power due to 1.2 times realistic decay heat is 0.00021 and, for the ANS 1971 standard, it is stated to be 0.00089. (Both values are assuming the full core has been irradiated.) We do not understand the factor of four difference in these values. Please explain.

    DBNPS Response

    The difference is in the assumptions for the operating life times. Using the infinite operation required by 10 CFR 50, Appendix K, the decay heat fraction was calculated to be 0.00089. Based on a more realistic 800-day operating history, the decay heat is much lower and results in a normalized fraction of 0.00021.

30. Please discuss mitigation of an RCS leak that is within the makeup capacity of the charging pumps from the viewpoint of the BWST inventory depletion and the unique conditions that may exist during the proposed operation.

DBNPS Response

A leak within MU capacity is one that at NOP is approximately 300 gpm or less with two MU pumps running. The direction provided in DB-OP-02522, Small RCS Leaks, will address this situation. Leak isolation would be attempted and a plant cooldown/depressurization commenced. Greater than 24 hours would be available before the BWST would be depleted (based on BWST volume of 500,000 gallons). Additionally, the leakrate will decrease as RCS pressure is reduced. The plant would cooldown and depressurize to establish DHR operation or establish LPI sump recirculation if DHR operation is not possible.

33. Do instructions to open any RCS venting paths to accomplish depressurization include opening RCS letdown? If so, please discuss the effect on RCS inventory.

DBNPS Response

This is not recommended because RCS inventory will be passed outside containment into the MU tank.

36. The FANP Page 38 discussion of the IMI nozzle break appears to assume one break cannot cause subsequent breaks in other IMI lines due to impingement effects. Is this correct? If so, please substantiate the assumption.

DBNPS Response

The break of a single IMI nozzle on the outside diameter (OD) that results in an impingement on a second nozzle that breaks on the inside diameter (ID) can be mitigated with acceptable consequences. Additionally, the break of a single IMI nozzle on the ID that results in an impingement on up to three other nozzles causing ID breaks can be mitigated with acceptable consequences.

37. In several locations, FANP states that adequate core cooling is assured once LPI flow is established following throttling of HPI flow to achieve RCS depressurization to below LPI flow initiation pressure. Please discuss this conclusion with respect to having to throttle flow to reduce pressure, but then a potential increase in flow rate is not a pressure problem.

DBNPS Response

The RCS can repressurize if the energy addition rate (core decay heat plus ECCS injection flow) into the system exceeds the break energy discharge. At the low core decay heat levels for this test, any break size that is a LOCA will easily remove all the core-generated energy.

As the LPI begins to flow, the system will approach a quasi-steady pressure when the energy flows remain in balance.

39. Page 1 of the commitment list states that "if IMI nozzle leakage is discovered, the proposed exception would not be utilized for Mode 3 entry following corrective action." Does this mean the HPI hardware problem would be resolved prior to Mode 3 entry? Please also address Mode 4 entry.

    DBNPS Response

    Please see the responses to earlier Question 3 and Question 8.

40. Page 2 of the commitment list indicates a cooldown to at least Mode 4 in case of equipment inoperability. Does this mean there is no potential need for ECCS in Mode 4? Please discuss.

    DBNPS Response

    The DBNPS TS 3/4.5.2 is applicable only in Modes 1, 2, and 3. The DBNPS TS do not currently require HPI operability in Mode 4. Without this commitment, the DBNPS would be allowed to remain in Mode 3, using the exception, with inoperable systems and components that are important in reducing the risk associated with the inability of the HPI pumps to maintain suction from the containment emergency sump (via the LPI pumps), provided the appropriate corrective actions are taken as provided in the TS associated with the inoperable equipment. The commitment would result in removing the unit from the mode of applicability in which the exception is necessary. Transitions from the DBNPS safe shutdown state (Mode 3) to Mode 5 are performed using non-safety grade components. 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.

Docket Number 50-346
License Number NPF-3
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Enclosure 2

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NUMBER 1 (DBNPS) LICENSE AMENDMENT APPLICATION TO REVISE TECHNICAL SPECIFICATION (TS) 3/4.5.2, EMERGENCY CORE COOLING SYSTEMS – ECCS SUBSYSTEMS – $T_{avg} \geq 280°F$ (LICENSE AMENDMENT REQUEST NO. 03-0008)

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

(29 pages follow)

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

1. In effect, FENOC states on page 2 of its cover letter that operation of the High Pressure Injection (HPI) pumps prior to piggy-back mode is not of concern. Has FENOC addressed the cleanliness of the Borated Water Storage Tank (BWST) water and the potential for stirring up debris in the Borated Water Storage Tank during initial HPI pump operation, such as due to return of bypass water to the tank? If so, please describe FENOC's assessment. If not, please provide an assessment of the cleanliness of the BWST water with respect to potential debris that may be of concern to HPI pump operability.

   DBNPS Response

   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. As clarified by the NRC staff technical reviewer, this question is only applicable for the use of modified HPI pumps with internal strainers that have not been completely qualified.

   The BWST is required by Technical Specification 3.5.4.a to have a minimum available inventory of 500,100 gallons during operation in Modes 1 through 4. The suction line for the BWST extends approximately 4 inches above the bottom of the BWST. This arrangement makes it unlikely that debris, if any exists, on the bottom of the BWST would be drawn into the HPI and Low Pressure Injection (LPI) pumps. To provide additional confidence in the quality of the water contained in the BWST, prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008, 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 Reactor Coolant System (RCS) as follows:

   1. 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.
   2. 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.

   Since the maximum flow rate of the HPI pump minimum flow recirculation is approximately 57 gpm per pump, the testing is expected to result in flow into the BWST at a flow rate that will agitate the BWST contents to a greater extent than that provided by the HPI pump

minimum flow recirculation during accident conditions. Although the flow rate out of the BWST is less than would be expected for some accident conditions that result in initiation of HPI, LPI and Containment Spray, the recirculation flow is expected to provide adequate agitation and to provide a conservative sample.

5. Have all locations where debris is of potential concern in the HPI trains been confirmed to be free of all debris? In the Low Pressure Injection (LPI) trains and the cross-over piping leading to the HPI pump suctions? In the containment emergency sump? Please provide information relative to confirming that there is no identifiable debris in containment or in the sump immediately prior to leaving Mode 5 to progress to Mode 3.

   DBNPS Response

   As clarified by the NRC staff technical reviewer, this question is only applicable for the use of modified HPI pumps with internal strainers that have not been completely qualified.

   In addition to the BWST sampling discussed in the Question 1 response, prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008, if the HPI pumps are modified to incorporate internal strainers, the piping associated with HPI pump operation will be flushed as follows:

   1. 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.
   2. 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.

   Flushing the piping associated with HPI pump operation at these flow rates is expected to ensure that there is no debris in the piping that could result in clogging the HPI pump strainers.

   In addition, the section of piping between the containment emergency sump and valves DH9A and DH9B will be inspected for debris prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008.

   The proposed amendment (DBNPS letter Serial Number 2950) would allow a limited entry into Mode 3 with the two HPI pumps potentially susceptible to damage from debris small enough to pass through the containment emergency sump screen, if required to operate in

tandem with the LPI pumps aligned to the containment emergency sump following a loss-of-coolant accident (LOCA). Although containment sump cleanliness is not relevant to operation of the HPI pumps during operation under the limited exception in Mode 3 it is relevant to LPI pump operation, and containment emergency sump inspections are required by DBNPS Technical Specification (TS) Surveillance Requirement (SR) 4.5.2.c and SR 4.5.2.d.2.a. These surveillances are required to be performed during plant heatup prior to entering Mode 3.

11. We need additional information in regard to Reactor Coolant System (RCS) heat loss. The following apply:

    a. Please estimate the RCS heat loss rate assuming the RCS water is at 532°F. Include a breakdown by major RCS components.

    DBNPS Response

    The reactor coolant system at the DBNPS is primarily insulated with reflective mirror type insulation. In general, the design insulation heat losses at full temperature vary between approximately 55 BTU/hr/ ft^2 (piping) to approximately 120 BTU/hr/ft^2 for the reactor vessel head. Original RCS design heat losses for various components that are directly connected to the RCS (as provided by Babock and Wilcox) are provided in the table at the end of this question, which indicates the total original design heat loss for primary components at full temperature was approximately 461 kW. However, the actual heat loss rates are believed to be much higher than these values and will vary over the course of plant life with the condition of the insulation, in part because insulation performance can be greatly reduced by convective losses between the panels and at vessel penetrations. Also, the heat losses at less than full temperature have not been estimated. However, operating data can be utilized to understand the relative value of ambient loss compared to current decay heat.

    On May 20, 2003, a small heatup of the Reactor Coolant System was attempted to support HPI pump testing and a 250 psig pressure test while still in mode 5. With the steam generators in wet layup, decay heat coolers were bypassed to begin a small heatup. RCS temperature increased from approximately 93°F at an initial rate of only approximately 19.7°F per day. By the afternoon of May 22, 2003, RCS temperature had had begun to plateau at approximately 120°F. A small additional increase to approximately 125°F was attained by spraying the pressurizer for oxygen control, which by causing an outflow, effectively utilized pressurizer heaters to heat up the system. However, the system temperature could not be increased further without use of reactor coolant pump heat. It must be noted that in this example, heat losses would occur from the entire RCS as well as the decay heat system loop. Additionally, to facilitate inspections, normal insulation was not in place on many small nozzles nor on the reactor coolant pumps. However, it is judged that even with the insulation intact, ambient losses from the reactor coolant system alone would balance decay heat well below normal system operating temperature.

Additionally, following a LOCA, the containment vessel water level is calculated to be at an approximate elevation of 566.87' to 570.77'. At this elevation, the exterior of the reactor vessel is submerged 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 reactor vessel. The annular space between vessel and insulation would fill to approximately the same level as the containment vessel. It would not be displaced by steam pressure due to vent paths between numerous insulation panels, or due to direct lifting of inspection panels. This would provide effective heat transfer through boiling. FENOC believes that submergence of the vessel could, by itself, provide adequate core cooling if required to do so during the planned Mode 3 NOP test.

Original design heat loss table at full temperature by component:

Component	Loss (BTU/hr)	Loss (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 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

b. FENOC, in 4.1.3 on Page 10, states "Due to this low reactor core decay heat generation rate, the combination of core inlet subcooling and heat losses through the core barrel wall and other structures should be sufficient to prevent the core boron concentration from exceeding the solubility limit." Why isn't a more definitive statement provided? And if heat is escaping from the core via the core barrel, what is the subcooling at the core entrance and what is the heat loss from the Reactor Vessel (RV)?

DBNPS Response

Boron concentration control analyses were not performed for this Mode 3 low decay heat test, therefore, a more definitive statement was not provided because the information was qualitative. The actual core decay heat level is extremely low from the long shutdown period (more than 17 months) and replacement of one-third of the reactor core with fresh unirradiated fuel. Heat losses should be sufficient to limit core boiling such that core concentrations could not reach the solubility limit at any RCS pressure, i.e., corresponding saturation temperature. Boron can concentrate in the core following a relatively large loss-of-coolant accident (LOCA) in the cold leg pump discharge (CLPD)

region. However, there is insufficient boron in the RCS and BWST for the core to reach the solubility limit at pressures above 92 psia, which is the maximum pressure for which boron precipitation control is required. At these pressures there will be high LPI flows that will enter the downcomer, mix with reactor vessel vent valve (RVVV) steam, and absorb heat added to the downcomer fluid from the core barrel or reactor vessel shell before spilling out of the large hole in the CLPD piping. With little core decay heat, the LPI injection temperature will quickly approach the temperature of its cooling water source; approximately 120°F. This liquid temperature will provide a temperature gradient of approximately 100°F, assuming saturation at atmospheric conditions, or more across the core barrel wall that will remove heat from the core region. This heat transfer will suppress or eliminate core boiling, and therefore reduce or eliminate concentration of boric acid.

c. The analyses apparently do not include RCS heat loss rate. If this is correct, please assess the influence of heat loss on the conclusions.

DBNPS Response

Heat losses were not credited in the FANP evaluation. Heat losses are traditionally not credited in short term 10 CFR 50.46 analyses because they are small in comparison to the core decay heat power. The heat losses were also conservatively not included in the post-LOCA boron precipitation analyses because it reduces the boiling (the concentrating mechanism) in the core region. Taking credit for the RCS heat loss would contribute little for full power analyses, but is a significant benefit for these low power conditions.

13. Please address the range of Incore Monitoring Instrumentation (IMI) breaks that cannot be mitigated under full power conditions. Under the conditions representative of the FENOC request.

DBNPS Response

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.

14. Is the completion of paper work the only thing keeping the pilot-operated relief valve (PORV) from being safety-related? (4.1.3, Page 8)

DBNPS Response

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 13RFO 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.

Of note is Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," in which the NRC determined that for operating pressurized water reactor plants it was not cost effective to replace existing non-safety grade PORVs and associated control systems with PORVs that are safety-grade even when they have been determined to perform safety-related functions, as discussed in the generic letter. In a safety evaluation dated August 22, 1994, the NRC found that the responses for the DBNPS either met the intent of the generic letter or were acceptable.

15. FENOC has proposed changing the emergency operating procedures (EOPs) to allow terminating HPI (and presumably Makeup (MU)) to accomplish RCS depressurization when subcooling margin (SCM) does not exist. This is a reversal of the fundamental EOP philosophy that injection shall continue under lack-of-SCM conditions. Consequently, please contrast the benefits and weaknesses of the FENOC approach with the existing EOPs where injection would be required. Specifically, the staff requests a comparison of using the HPI pumps to fulfill the injection requirements as contrasted to FENOC's approach of throttling or terminating HPI with a philosophy that HPI could be restarted if the desired results were not obtained. Please also address the potential for future misunderstanding in applying EOPs when the SCM requirement will apply. A similar comparison should be provided regarding the temporary change in LPI flow rate that would allow termination of HPI.

DBNPS Response

It is important to note that throttling of HPI without subcooling margin in response to certain SBLOCA is a licensing basis option under the proposed amendment, but it is not the first choice of several options that would be available depending on equipment availability and specific conditions. The first choice method is to depressurize the RCS using reactor coolant pumps and steam generators. If this method is not achievable, then the RCS would be depressurized using the PORV and high point vents. Only if this second method is not available would the need to throttle HPI without subcooling margin be used.

Many factors are different for this Mode 3 NOP test from the traditional accident assumptions:

1) The decay heat and boiling rates are extremely low compared to normal post accident assumptions.
2) Although not credited in short-term 10 CFR 50.46 analyses, ambient losses alone would be expected to be a significant influence leading to post accident cooldown.
3) RCS pressure is held up by pressure balance between injection and break flow for small breaks that do not depressurize themselves, rather than by the decay heat contribution.
4) The effective flow area of the PORV is sufficient to reduce RCS pressure to where LPI can inject prior to expending the BWST.
5) Incore thermocouples are expected to provide reliable indication of core heatup before fuel damage could occur.
6) Throttling of the HPI flow is based on the normal operating pressure (NOP) test conditions. Should the operator throttle too quickly for the existing condition, this can be corrected by re-initiating sufficient flow for the condition.
7) The RCS was recently pressurized to 250 psig and inspections performed to identify leakage. Inspections of the RCS pressure boundary will be performed during the NOP test and are expected to provide early indication of leakage.

8) The Technical Support Center (TSC) would be manned (within approximately 1 hour) before throttling of MU flow and HPI flow to reduce pressure would be required (at approximately 2 hours), providing additional resources for determining the appropriate actions.

9) Offsite power sources are not expected to be perturbed by transfers due to reactor trip.

10) Post-accident access to all areas outside containment would be greatly improved due to the long period of time already provided for the decay of fission products.

For small RCS break sizes (such as an IMI nozzle break or partial crack), the HPI will be able to meet and exceed the break flow rate once activated. The excess HPI will refill the RCS liquid inventory that was lost and, once refilled, the RCS will repressurize until the break and HPI flow are in equilibrium. This equilibrium pressure will not be perturbed unless the Emergency Core Cooling System (ECCS) injection rate is decreased. Throttling the HPI flow to reduce RCS pressure to within the LPI operating pressure range is an action unique to this Mode 3 NOP test.

Please see the response to Question 42 for details of the expected changes to the EOP. 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 NOP test. 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 NOP test.

16. Are RCS vents safety grade? Fully operable under the proposed Mode 3 conditions?

DBNPS Response

The RCS high point vents are safety grade. The vent on each hot leg is controlled by two solenoid-operated valves. The valves are Nuclear Class 1 and are seismically and environmentally qualified. The valves are powered from Class 1E power supplies. These valves are required to be operable in Mode 3 by the DBNPS Technical Requirements Manual requirement 3.4.11. The high point vents have a thick-edged orifice with an internal diameter of 0.19 inches (0.0002 ft^2 per vent). The Moody critical mass and energy flow from each vent (ignoring the line losses) is approximated in the Figures 1 and 2. Use of the high point vents will assist in depressurizing the RCS. Their effectiveness is based on the conditions under which they are opened. However, they are not big enough to depressurize the RCS to the LPI pressure range under all conditions. Therefore, although their use is endorsed to help depressurize the system, reliance upon them is not credited in the evaluations.

Figure 1. High Point Vent Mass Flows Per Vent

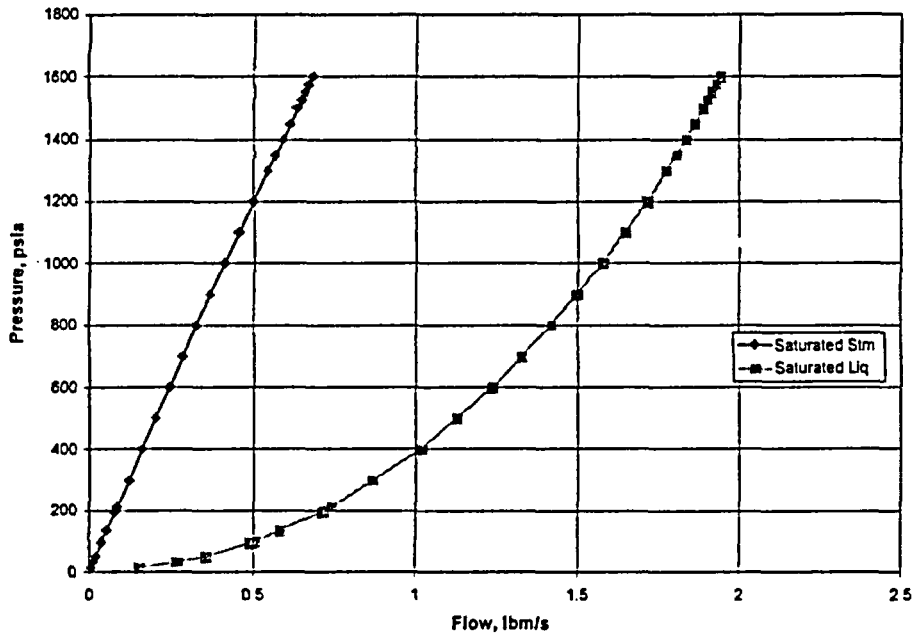
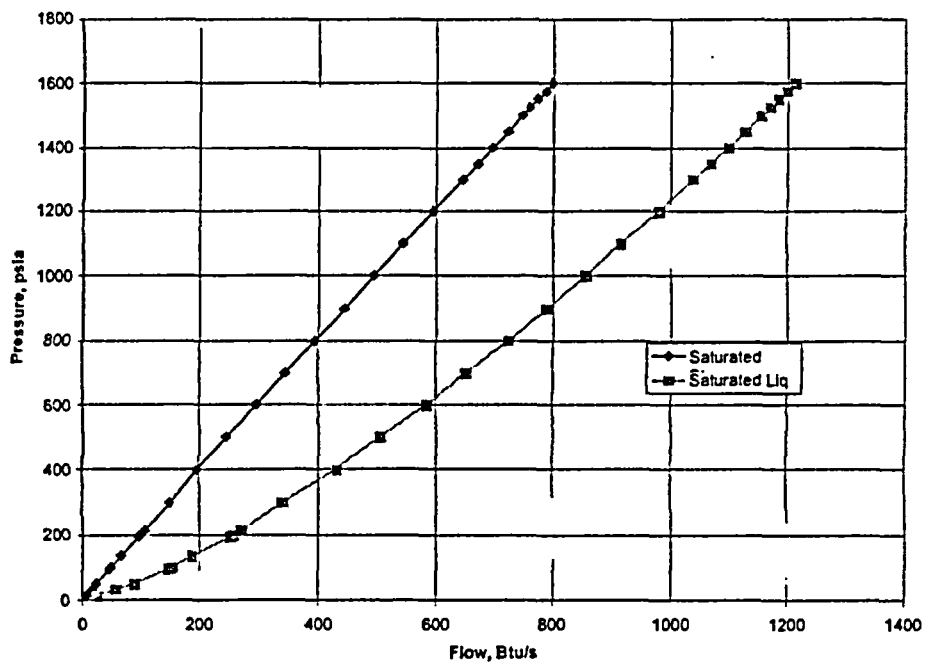


Figure 2. High Point Vent Energy Flows Per Vent



17. With respect to auxiliary spray:

a. FENOC identifies that LPI flow through the auxiliary pressurizer spray (APS) line is an available redundant flow path to provide water to the RCS. Please provide an estimate of when such flow would be initiated, when water would begin to reach the RV, and when any core heatup would be effectively addressed. Provide maximum temperatures for this scenario. (4.1.3, Page 10)

DBNPS Response

The ECCS was designed to provide multiple water sources and multiple injection locations to assure that adequate flow will be provided to the RCS. Considering the scenario of a broken injection line as the accident initiator, and a single failure that incapacitates one train of pumped injection, adequate injection flow must be provided to remove core decay heat. The break of relevance to this LAR and use of the auxiliary pressurizer spray (APS) line is a double-ended break of the core flood line. This break would be initially mitigated by HPI with no change relative to this LAR. However, conservatively assuming loss of HPI once sump recirculation begins and unavailability of the LPI cross-tie line, the alternate flow path through the APS would be used. Because of the relatively large break area associated with the core flood line break, approximately 0.44 ft^2, the RCS pressure will be below the LPI shut-off head within minutes.

Initiation of APS should begin after initiation of emergency sump recirculation if ECCS injection is only provided to one location. Opening the LPI cross-tie is preferred if possible. If not available, HPI should be kept running if possible. If no HPI is available, the LPI could also be injected into the pressurizer through 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, core cooling would begin after the pressurizer was filled (in approximately 5 hours). Core heat up would be dependent upon the scenario.

For a break in the core flood line, the intact core flood tank (CFT) flow and the HPI flow during the BWST draining period would refill the RV to the break elevation. In this case there should be approximately 6.5 ft of water between the CFT nozzle and the top of the core at the time of sump switchover. The fluid flow area in the RV in the core region is approximately 100 ft^2. The flow area is slightly larger above the core region, however using 100 ft^2 is conservative. The liquid mass in the approximately 650 ft^3 above the top of the core is approximately 38900 lbm (650 ft^3 / 0.016719 ft^3/lbm) at saturated conditions at 14.7 psia. This liquid could absorb approximately 37.7×10^6 Btu (38900 lbm * 970.3 Btu/lbm) in being boiled off to steam. For a decay heat level of 1360 Btu/s the liquid could support approximately 7.7 hours (37.7×10^6 Btu / 1360 Btu/s / 3600 s/hr) before the core would uncover without any ECCS injection. Therefore, as discussed above core cooling with APS would begin before the core would uncover.

b. Is it possible that APS would be in use at the same time the pressurizer PORV is open? Or for the case of a pressurizer break? If so, please discuss the effectiveness of APS.

DBNPS Response

The APS should not be initiated when the PORV is open because it can bypass the APS flow. If the PORV is open or if the break is in the pressurizer, there is a single pass cooling arrangement that allows the ECCS to flow through the core and out the break. Core exit subcooling will be restored and no actively initiated post-LOCA boron concentration control is needed.

18. The discussion of boron precipitation control (BPC) during long-term cooling does not address certain phenomena that are of concern to the staff. Specifically, in a boiling system, there will be a temperature gradient due to elevation and precipitation that can occur at the top of the liquid prior to occurrence at lower elevations. Further, if there is significant concentration where credit is taken for the increased boric acid solubility due to elevated temperature, how is the RCS later cooled down without a precipitation concern? Please address these concerns (4.1.3, Page 10)

DBNPS Response

Calculations were performed to estimate how long it would take to boil a sufficient amount of liquid to concentrate the boron in the core. Without crediting heat losses, it would take approximately one week of the current decay heat to provide that integrated power (i.e., sufficient boiling) to achieve the solubility limit at 14.7 psia. It will take much longer if the RCS is pressurized. The EOPs instruct the operators to initiate the active boron concentration control methods shortly after the switchover to emergency sump recirculation. This time is well before the core will approach boron concentrations that could result in precipitation. BPC is initiated when: 1) adequate Subcooling Margin 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.

19. What are the environmental conditions and estimated times associated with opening the LPI cross-tie line and with respect to initiating LPI flow through the auxiliary pressurizer spray line? (4.1.3, Page 11)

DBNPS Response

The LPI train cross-connect valves are normally operated from the control room. In the event of a loss of an 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 auxiliary pressurizer spray 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 conservatively expects that the valves could be repositioned within 30 minutes of the decision to initiate LPI flow through the auxiliary pressurizer spray 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 auxiliary pressurizer spray line are performed to be accessible following a LOCA (specifically a CFT line break) as the Framatome-ANP (FANP) evaluation shows that no fuel damage is expected to occur.

21. What is the equilibrium pressure and flow with the PORV open and LPI injection ongoing?

DBNPS Response

The PORV area, based on a bore diameter of 1.62 inches, is 0.0143 ft^2. At an RCS pressure of 200 psia, 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 psia, which is within a few psi of the LPI pump shutoff head. This calculation is considered to be conservative since the flow through the break is not accounted for.

24. FENOC discusses maximizing the availability of plant systems and components. It also states that required surveillance testing will continue to be performed and that activities in the offsite power switchyard and electrical switchgear rooms will be limited to those of an essential nature. Please address the specific activities that would still be conducted and explain why they cannot be eliminated by planning. (5.2, Page 19)

DBNPS Response

In the May 14, 2003, submittal (DBNPS letter Serial Number 2950) FENOC indicated that although maintenance activities that adversely affect operability will not be performed, required surveillance testing will continue to be performed on the Low Pressure Injection System, Decay Heat Removal System, Emergency Diesel Generators, Auxiliary Feedwater System, the Motor-Driven Feedwater Pump, Steam Generator Atmospheric Vent Valves, the Pressurizer Pilot-Operated Relief Valve, RCS Hot Leg High Point Vent Valves, Pressurizer High Point Vent Valves, the Hot Leg Level Monitoring instrumentation, incore thermocouples, and any necessary support systems and electrical power sources. A review of the current outage schedule for the Mode 3 NOP test shows that the following surveillance testing will be performed on the listed systems/components and could affect the operability of the system/component during the test.

- The CFT outlet check valves (CF28 and CF29) reverse flow tests are performed with RCS pressure greater than 700 psig and are required by the IST Program and TS 4.0.5.

Performance of the reverse flow tests renders the associated LPI train inoperable during the test.

- The HPI back-to-back check valve (HP57 &HP59, HP56 & HP58, HP48 & HP50, and HP49 & HP51) isolation integrity test is performed in Mode 3 and is required by a commitment in the DBNPS response to Generic Letter 87-06, "Periodic Verification of Leak Integrity of Pressure Isolation Valves," and by TS 4.0.5. During performance of the test, one HPI train at a time is rendered inoperable, one boron injection flow path at a time may be rendered inoperable, and the normal MU flow path will be isolated. Individuals are assigned to restore each inoperable HPI train, each boron injection flow path, and the normal MU flow path.
- Auxiliary feedwater pump and valve surveillance testing is performed in Mode 1, 2, and 3 as required by TS 4.7.1.2.1.a.1, TS 4.7.1.2.1.c.2, the IST Program and TS 4.0.5. Both auxiliary feedwater trains are required to be tested during the Mode 3 RCS leak test. During the surveillance testing, each train will periodically be incapable of performing its safety function. The operators are instructed that if the system is required to operate for its safety function, the operators performing the test are to restore the system to normal.
- The RCS isolation check valves (CF30, CF31, DH76, and DH77) leak tests are performed with the plant at normal operating pressure (NOP). The testing is required by TS 4.4.6.2.2 and TS 4.0.5 (IST Program). During the performance of this test the respective LPI flow path will be rendered inoperable.
- The RCS resistance temperature detector (RTD) cross calibration is performed with the plant in Mode 3. The testing is required to satisfy the Limiting Conditions for Operations for TS 3.3.1.1, TS 3.3.3.5.1, and TS 3.3.3.6. Performance of this calibration results in the inoperability of hot leg RCS temperature instrumentation, RCS loop outlet temperature instrumentation, and RCS hot leg level (wide range) instrumentation at various times during performance.
- In addition, the pressurizer PORV and its associated block valve are stroke tested in Mode 3. This testing is required by the IST Program, TS 4.0.5 and TS 4.3.3.5.2.

Additions to the outage schedule are reviewed by the Scope Review Team prior to incorporation in the 13RFO 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 NOP test.

25. FENOC states that an additional dedicated licensed operator (above the minimum TS manning requirement) will be on shift in the control room. What is the normal complement in the control room when progressing from Mode 5 into the upper operating temperature and pressure typically associated with Mode 3? (5.2, Page 19)

DBNPS Response

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 (who may be one of the two licensed Senior Reactor Operators). 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 are more restrictive than the TS and 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 (SRO), the Shift Engineer (STA), the Unit Supervisor (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 3 and Mode 4 for the NOP test.

29. We note the FANP correction for inert fuel is to multiply the decay heat by 101/177. Since the fuel being removed probably has the highest irradiation, does this result in an under-prediction of the decay heat generation rate? If so, by how much?

DBNPS Response

The radial power distributions for three different assumed reload power distributions are shown below for a 72 feed batch size (Same batch size as in Cycle 13). The information from the assumed representative peaking contributions suggests that the previously twice-burned fuel is near the core average power. The current DBNPS core now has 76 fresh fuel assemblies. For the example given, 72 once-burned fuel assemblies are reinserted with 29 twice-burned fuel assemblies. The 72 once-burned assemblies were run at a radial power greater than unity, however, their time at operation is limited to one cycle. For once burned fuel assemblies, the total decay heat contribution considering the effect of days at operation preceding 1.5 years post-trip is substantially less than the decay heat contribution considering the effect of the radial power factor from the first-burn. The twice-burned fuel assemblies have a larger time at operation, but the average power for the first two burns is lower. The average radial power with the twice-burned days at operation is also substantially below the infinite operation curve. Therefore, the adjustment of the infinite operation decay heat curve by the 101/177 is reasonable without rigorous analysis of the actual core decay heat. This is

especially true given that the 120% decay heat factor is included in the analyses for 1.5 years post-trip.

Example	First Cycle	Second Cycle	Third Cycle
1) Radial Power Distribution	1.25	1.02	0.41
Number of Assemblies	72	72	33
Core Power Fraction	0.508	0.415	0.076
2) Radial Power Distribution	1.135	1.135	0.41
Number of Assemblies	72	72	33
Core Power Fraction	0.462	0.462	0.076
3) Radial Power Distribution	1.2	1.07	0.41
Number of Assemblies	72	72	33
Core Power Fraction	0.488	0.435	0.076

32. The FANP discussion appears to focus on one train of ECCS (or one HPI) being in operation. (See, for example, the last line of Page 25, although there are many examples in the following pages.) Is this correct? If so, do two trains exacerbate the potential concern? See also our question pertaining to MU pump operation.

DBNPS Response

The discussions in Section 4 of the FANP report are based on the minimum ECCS analysis flows for a single HPI and LPI pump. Accordingly, Figures A-2 and A-3 of the FANP report give the pressure equilibrium conditions with one train of ECCS in operation. If two trains of ECCS are in operation, the break area will be double the area given in Figures A-2 and A-3. It would also be increased by any MU flow that may actually be available. When the total injection flow rate is increased, the RCS will equilibrate at a higher pressure. Consideration of the maximum ECCS flows, including two MU pumps, was included in the throttling guidance in Section 5.2 of the FANP report. 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.

38. The FANP Page 44 discussion indicates a 3.5 percent head reduction was assumed for HPI flow rate. Is this a conservative assumption when the concern involves HPI flow causing RCS pressure to remain high?

DBNPS Response

The response to this question is an extension of the response to Question 32. If the head flow curve is marginally higher, the results will be marginally different in terms of RCS pressure

response and break area equilibrium points. The head reduction does not alter the validity or the effectiveness of the throttling guidance.

41. Please address training of the affected Technical Support Center staff with respect to the Emergency Operating Procedure changes relative to this limited exception.

DBNPS Response

Potentially affected Technical Support Center (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 NOP test. 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 NOP test.

42. Provide a summary of Emergency Operating Procedure changes that will be made relative to this limited exception.

DBNPS Response

The following response provides an overall summary of the expected changes that will be made to the Emergency Operating Procedure (EOP) for the Mode 3 NOP test. They may be revised as comments are received during the procedure change review process and the EOP validation process.

A step will be added early in Sections 10, 11, 12, and/or 13 of DBNPS Emergency Operating Procedure DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture. The following sections are accessed following initial symptom mitigation:

10.0 A Large LOCA has Occurred and the Core Flood Tanks are Emptying

11.0 Transient Termination Following an Occurrence that Leaves the RCS Saturated with SG(s) Removing Heat

12.0 Transient Termination Following an Occurrence that Leaves the RCS being Cooled by MU\HPI Cooling

13.0 Transient Termination Following an Occurrence that may Require Pressurizer Recovery or Solid Plant Cooldown with SG(s) Removing Heat and RCS Subcooled

This step provides a reference to a new attachment that will be added to DB-OP-02000 to provide direction based on the guidance provided by the FANP report. If a LOCA occurs during the Mode 3 or 4 period, the instructions of this attachment will be applicable. The direction will be based on the plant status for the following plant conditions. A description of the conditions with an overview of the actions for each condition follows:

Condition 1 Small Break LOCA (SBLOCA), where Plant Stabilizes with Adequate SCM and RCS Pressure is Greater than LPI Shutoff Head.

Initiate cooldown and depressurization of the RCS. This would most likely involve forced circulation of the RCS, but natural circulation would also be effective if Reactor Coolant Pumps are not available.

Core heat plus the heat removed for cooldown would be transferred to the Steam Generators (SGs). Heat in the SGs would be sent to the condenser if available or to atmosphere via the atmospheric vent valves if the condenser is not available.

Secondary inventory would be provided by either of the two Main Feedwater Pumps, by either of the two Auxiliary Feedwater Pumps, by the Motor-Driven Feedwater Pump, or by the Startup Feedwater Pump.

Primary pressure would be reduced by turning off the pressurizer heaters and allowing losses to ambient supplemented by pressurizer spray (with forced circulation) and/or venting the pressurizer. Throttling of MU and HPI flow would be conducted per the current procedural guidance.

Primary inventory would be maintained by the either of the MU pumps and/or the HPI pumps taking suction from the BWST.

Cooldown and depressurization would progress until the RCS reaches approximately 200 psig. With adequate SCM, it is expected that SG heat transfer would allow RCS cooldown prior to depleting the BWST. At approximately 200 psig, one train of LPI train would then be put in the Decay Heat Removal mode while the remaining LPI Pump stays in the injection mode.

If unable to establish two injection flow paths via LPI, establish auxiliary pressurizer spray flow to create a second injection flow path.

When BWST level reaches approximately 9 feet, ECCS suction would be manually transferred to the emergency sump. At least one LPI train would be maintained taking a suction on the emergency sump.

Condition 2 Small Break LOCA (SBLOCA), where Plant Stabilizes with Inadequate SCM and RCS Pressure is Greater than LPI Shutoff Head.

Initiate cooldown and depressurization of the RCS. Forced circulation of the RCS would not be permitted with a loss of adequate subcooling margin. Single-phase natural circulation would not be available and continuously effective boiler/condenser circulation may be unsustainable due to very low decay heat.

Core heat plus the heat removal for cooldown would be directly removed by the break flow and some heat would be transferred to the SGs. Heat in the SGs would be sent to the condenser if available or to atmosphere via the atmospheric vent valves if the condenser is not available.

Secondary inventory would be provided by either of the two Main Feedwater Pumps, by either of the two Auxiliary Feedwater Pumps, by the Motor Driven Feedwater Pump, or by the Startup Feedwater Pump. However, secondary heat transfer is not needed to remove core energy.

Primary pressure would ride the saturation line unless SCM is restored as cooling and injection flows increased.

Primary inventory would be maintained by the MU Pumps and/or the HPI pumps taking suction from the BWST. Cooldown and depressurization would progress until the RCS reaches approximately 200 psig with a goal of establishing LPI flow to the RCS prior to transferring ECCS suction to the emergency sump.

Continuously monitor cooldown rates to ensure that LPI pump shutoff head pressure will be reached prior to expending the BWST.

If unable to reach an RCS pressure that would allow LPI flow to the core, then other options available to reduce RCS pressure to allow LPI injection flow prior to sump transfer include the following, in order of preference:

- Opening pressurizer and RCS high point vents,
- Opening the PORV (APS will not be used if the PORV is open),
- Throttling MU flow without adequate SCM in accordance with the FANP guidance, or
- Throttling HPI flow without adequate SCM in accordance with FANP guidance (This will include guidance to restore HPI flow if superheated incore thermocouple indications are observed).

Core Flood Tanks will be isolated during the depressurization to allow LPI flow and prevent injection of Nitrogen to the RCS.

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).

When BWST level reaches approximately 9 feet, ECCS suction would be manually transferred to the emergency sump. At least one LPI train would be maintained taking a

suction on the emergency sump and providing a suction source to HPI if necessary or to inject to the RCS if conditions permit.

Condition 3 Small Break LOCA (SBLOCA), where Plant has Inadequate SCM and RCS Pressure is Less than LPI Shutoff Head.

Initiate cooldown and depressurization of the RCS. Forced circulation of the RCS would not be permitted with a loss of adequate subcooling margin. Single-phase natural circulation would not be available and continuously effective boiler/condenser circulation may be unsustainable due to very little decay heat.

Core heat plus the heat removal for cooldown would be directly removed by break flow and some heat would be transferred to the SGs. Heat in the SGs would be sent to the condenser if available or to atmosphere via the atmospheric vent valves if the condenser is not available. RCS cooling due to break flow alone is expected to allow the RCS pressure to reach LPI injection pressure.

Secondary inventory would be provided by either of the two Main Feedwater Pumps, by either of the two Auxiliary Feedwater Pumps, by the Motor Driven Feedwater Pump, or by the Startup Feedwater Pump. However, secondary heat transfer is not needed to remove core energy.

Secondary inventory would be provided by either of the two Main Feedwater Pumps, by either of the two Auxiliary Feedwater Pumps, by the Motor Driven Feedwater Pump, or by the Startup Feedwater Pump.

Primary pressure would ride the saturation line unless SCM is restored as cooling and injection flows increased.

Isolate the CFTs prior to expending the liquid inventory.

Primary inventory would be maintained by the MU Pumps and/or the HPI pumps taking suction from the BWST. Cooldown and depressurization would progress until the RCS reaches approximately 200 psig with a goal of establishing LPI flow to the RCS prior to transferring ECCS suction to the emergency sump. When LPI flow is established, the MU pumps will be shutdown.

APS would be initiated from LPI for boric acid precipitation control.

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).

Condition 4 Large Break LOCA where Plant has Inadequate SCM and Core is Being Cooled by LPI.

In the large break LOCA scenario, RCS pressure lowers due to break flow.

The HPI shutdown criterion can be met allowing HPI to be removed from service prior to ECCS suction transfer to the emergency sump. 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).

APS would be initiated from LPI for boric acid precipitation control.

Steam Generator heat transfer would not be available or needed in this condition.

43. Provide docketed copy of the RCS pressure/time histories (full power as well as Mode 3) as supplied by FANP.

DBNPS Response

The RCS response to a LOCA from full power operation is different from the response to a LOCA from Mode 3 conditions. The response is highly break size dependent and is characterized in Section 5.2.3 of the DBNPS LOCA Summary report for the full power responses and in Section 4 of the DBNPS Mode 3 Low Decay Heat Evaluation (submitted as Enclosure 2 to DBNPS letter Serial number 2950). The responses were considered for the six liquid region LOCA break classification ranges typically used in the discussion of the full power analyses with minimum ECCS injection (i.e., 1 train – 1 HPI, 1 LPI, 1 Containment Spray). Break sizes that are within the makeup system capacity are considered leaks, not LOCAs. The leaks are generally less than 0.002 ft^2 and rely on operator actions to maximize makeup injection from one or two makeup pumps. The operators may also isolate letdown and reduce core power to manage the RCS during these leaks such that reactor trip will not occur within the first 10 minutes. These leaks rely on operator actions and equipment not traditionally credited in LOCA analyses for larger break areas.

The six categories of LOCA break sizes and their area ranges used in the full power analyses are generally characterized as:

1. SBLOCAs that may not interrupt natural circulation with MU pump flow (0.002 to 0.005 ft^2).
2. SBLOCAs that may allow the Reactor Coolant System (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. LBLOCAs (0.75 to 14.1 ft^2).

RCS Leaks

A LOCA is defined as any break size that is in excess of the makeup system capacity. This minimum break size is not strictly defined because it is dependent on break location, makeup and letdown flow rates, the critical flow model used in the analysis, plant equipment available, and operator actions that are credited with offsite power available. Accordingly, a variety of break areas can be given as the minimum break size for a LOCA based on different sets of inputs and different break locations. For DBNPS, a calculation was performed to define the minimum break size for a liquid region break. The calculation demonstrates that the leak rate for a ¾" Schedule 160 (A=0.00206 ft^2) line break would not initially be offset by normal operation of a single makeup pump. However, this break size did not depressurize the RCS to the low RCS pressure trip setpoint within 10 minutes. For this break to not be considered a LOCA, it is assumed that the pressurizer heaters function and the operator performs some or all of the following actions:

- isolate letdown,
- start a second makeup pump with the recirculation line closed,
- transfer suction from the makeup tank to the BWST
- trip the reactor, and
- initiate an orderly cooldown.

With these actions, the RCS remains in forced or natural circulation and the operators can use the steam generators to depressurize the system during the longer term phase of the event. The short term pressure response is shown in Figure 1 for leaks from full power initial conditions with high core decay heat levels.

Category 1: SBLOCAs Too Small to Interrupt Natural Circulation

Break areas larger than approximately 0.00206 ft^2 are considered a LOCA. These smallest LOCA break sizes are in excess of the makeup system flow delivery and will depressurize slowly and achieve a reactor trip within the first ten to twenty minutes following break opening. When MU pump flow is credited with some or all of the operator actions identified for leak mitigation, the RCS will evolve similarly to the leak. If the system loses core exit subcooling margin (LSCM) after the reactor trip, the RCPs will be manually tripped (per the EOPs) within two minutes of LSCM if they are not lost due to a loss of offsite power (LOOP). These smaller break sizes will not quickly depressurize to the low RCS Safety Features Actuation System (SFAS) trip pressure. As a result, the operators will have time to diagnose the symptoms of a LOCA (predominately LSCM or leakage greater than allowed

by Technical Specifications) and may maximize the MU pump flow. They may also manually activate SFAS. Once SFAS is initiated, HPI is actuated and letdown is isolated. The operators should also piggyback HPI on the LPI pumps if the RCS pressure remains high. With these described actions, the Category 1 breaks evolve similar to that for a leak. However, if the break is characterized as a LOCA, and only redundant safety-related equipment is used with limited credit for operator actions, the response for these tiny LOCAs can result in an interruption of natural circulation and higher RCS repressurization than would be noted for the Category 2 break.

After initiation of SFAS, the ECCS inflow can be capable of matching saturated liquid break flows for CLPD break sizes in the range of 0.002 to 0.0075 ft^2 depending on the RCS pressure, number of HPI pumps operating, and piggyback of the HPI pumps. If the ECCS injection (HPI plus MU) matches break flow, and auxiliary feedwater (AFW) flow is initiated either automatically or manually at a flow rate sufficient to remove the core decay heat not lost through break-HPI cooling, then the RCS will remain in single-phase natural circulation. The single-phase natural circulation flow provides a continuous core-to-steam generator energy transport mechanism that keeps the core from boiling and the RCS pressure well coupled to the secondary side pressure. As the system is depressurized with steam generator cooldown via the atmospheric vent valves, condenser (if available), or steam demand to the AFW pumps, the HPI flow will be throttled to maintain the desired core exit subcooling margin. These LOCA break sizes are easily mitigated by the combination of HPI, MU, and AFW flow. The HPI and MU compensates for break inventory loss, and the AFW provides core decay heat removal and system cooldown.

If the MU pump flow is not included, the HPI pumps would not provide sufficient ECCS injection at the higher RCS pressures to match the break flow. The system inventory loss would cause natural circulation to be interrupted. This interruption in flow would result in initiation of core boiling and RCS repressurization above the HPI shutoff head. When the RCS pressure is above the HPI shutoff head, the RCS liquid inventory decreases due to the break flow. The RCS inventory will cause the RCS levels to decrease. When the primary levels descend into SG tube bundle below the AFW injection location, a high-elevation boiler-condenser mode (BCM) of heat transfer is established in the SG. The SG heat removal would decrease RCS pressure well below the HPI shutoff head. The HPI flow would be reestablished and it would exceed the break flow for the smallest break sizes in this category. The HPI flow would refill the RCS above the AFW injection location and the SG heat transfer would be interrupted. The RCS would repressurize again above the HPI shutoff head and the cycle would be repeated unless natural circulation could be regained. The RCS pressure could cyclically oscillate between the secondary side pressure and 1800 to 1900 psia for the smaller breaks in this category. These oscillations could last for days or weeks with only a slight decrease in the peak pressure as the core decay heat rate decreased. Figure 1 shows this response labeled as Category 1-HPI.

Once natural circulation is lost, it is difficult to refill the RCS and reestablish single-phase natural circulation in at least one loop with flow from only the HPI pumps. If the MU pumps are available or the PORV is opened, the RCS can be refilled. Once refilled, the SG cooldown allows the RCS to be cooled to the conditions at which the LPI system can be initiated. Therefore, credit for flow from the makeup pumps is key in the response for Category 1. The RCS pressure for a Category 1 break with flow included from the MU pumps is shown as Category 1-MU in Figure 1.

Category 2: SBLOCAs that May Allow the RCS to Repressurize in a Saturated Condition

As the break liquid discharge increases above the ECCS inflow, inventory loss will cause the RCS to depressurize until the fluid in the hot legs saturate and begin to flash. The steam accumulation in the U-bend region will block natural circulation and interrupt the steam generator heat removal. For these LOCAs, with break areas ranging from approximately 0.005 to 0.035 ft^2, the steam generator removes core heat during the early portion of the transient, when the decay heat is high to prevent the RCS from repressurizing. When RCS liquid flow ceases, and the energy removal by the steam generator is interrupted, some repressurization will occur due to core boiling. The minimum RCS pressure reached prior to this repressurization will determine if the low RCS pressure SFAS trip is actuated. (If the trip is not achieved automatically, the operator is instructed by the EOPs to activate SFAS based on the loss of adequate subcooling margin.) The repressurization that occurs accelerates the rate of liquid loss out of the break if the break phase remains liquid only and reduces or prevents the HPI inflow if SFAS has been actuated and the system pressure is not too high. The repressurization is halted when the combination of break and steam generator heat removal matches or exceeds the core energy addition rate. Makeup injection is still very beneficial for this class of breaks, because the RCS pressure remains high. However, these break sizes in this range are included in the typical Evaluation Model analyses which do not credit MU pump flow to demonstrate acceptable core cooling. The Category 2 pressure response shown in Figure 1 does not include makeup pump flow.

The net loss of system liquid inventory will cause steam bubbles to form in the hot leg U-bends, and they can expand into the steam generator tube region. This expansion is established either by flashing of hot leg liquid during the depressurization periods or by an intermittent steam venting up the hot leg when the break discharge plus HPI condensation cannot offset all the core generated steam. If AFW is flowing when the level descends into the tube region and the primary pressure is greater than the secondary side pressure, high-elevation BCM will ensue. Condensation on the primary side will decrease RCS pressure. If the AFW is off because the secondary side has been refilled to the loss of subcooling margin level (above RCP spillover), then the BCM will be delayed until the primary level drops below the secondary side level. The secondary side pool level (pool) BCM will reduce RCS pressure before the vessel level has decreased below the bottom of the hot leg nozzle. With either the high-elevation or pool BCM, the core-to-steam generator heat removal mechanism is re-established. The heat transfer condenses RCS steam. This steam sink in combination

with the break and HPI reduce the RCS pressure to near that of the secondary side pressure. (It should be noted that if the SFAS trip setpoint has not been reached or the operators have not manually started HPI, this depressurization will initiate ECCS flow.) In some cases, the condensate can augment ECCS inflow by keeping the CLPS liquid full, such that liquid displaced by the condensate can flow over the pump into the reactor vessel.

Without any steam generator heat removal, the smallest Category 2 LOCAs could repressurize all the way to the pressurizer safety valve opening pressure, because the break energy removal is unable to relieve all the core-generated energy, through either liquid or steam discharge. At elevated RCS pressures, the HPI system will not be able to provide sufficient (or any) ECCS to make up for the core boiloff rate. With time, the RCS liquid inventory above the top of the core will be depleted and the core could uncover and heat up. This evolution, however, will not occur so long as AFW is preserved at a flow rate sufficient to remove the core decay heat, and the secondary side level is controlled to a level (the loss of subcooling margin level) that is above the RCP spillover elevation. In this configuration, a pool BCM will be established before the core uncovers. The pool BCM ensures that the RCS pressure can be controlled to a value slightly above the secondary side pressure. At these moderate RCS pressures, the HPI system can generally match core decay heat to prevent core uncovering. Should uncovering occur, HPI will limit the extent of the peak cladding temperature (PCT) increase.

The smaller Category 2 break sizes will not be able to depressurize the RCS below the secondary side pressure for many hours post-LOCA. The RCS pressure for these break sizes can be decreased via the steam demand of the turbine driven AFW pumps or via operator-initiated steam generator cooldown. This RCS cooldown could be interrupted if the RCS refills above the top of the tubes, thereby halting the high-elevation BCM. The cooldown can be continued when the RCS refills sufficiently to re-establish single-phase natural circulation, or when the subsequent RCS inventory loss causes the level to drop back into the tube region.

Category 3: SBLOCAs that Slowly Depressurize the RCS to Core Flood Tank (CFT) Pressure

As the break size increases, the break energy discharge to the reactor building replaces the steam generator as the primary core heat sink. For CLPD breaks in the range of 0.035 to 0.06 ft^2, the break energy discharge will exceed the core decay heat within a few seconds following reactor shutdown. The steam generator heat transfer via AFW is still important for these break sizes because it can condense RCS steam and augment the break in depressurizing 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.

These moderate break sizes limit the RCS depressurization rate shown in Figure 1. Generally, it takes 25 to 60 minutes (depending on break size, decay heat power, and steam generator 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 reactor vessel. The continuous HPI flow delivery to the vessel is most 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: SBLOCAs that Quickly Depressurize the RCS to the CFT Pressure

Break sizes from 0.06 to 0.25 ft^2 will depressurize the RCS to the CFT pressure within five to twenty-five minutes after break opening. The severity of the results somewhat depends on the total HPI flow delivery early during the transient. The CFT fill pressure is most important in the overall severity of results, because the CFT flow halts the core mixture level decrease and initiates vessel refill. Lower CFT pressures (nominal less operational band and uncertainty) delay the CFT refill which minimizes the core mixture level and maximizes the predicted PCT should core uncovering occur.

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 leg and steam generator tubes, and reducing the steam generator reverse heat transfer. Figure 1 shows the characteristic pressure response for the Category 4 break sizes.

Category 5: SBLOCAs that Depressurize the RCS Nearly to the Containment Pressure

Break sizes greater than 0.25 ft^2 (but less than 0.75 ft^2, the upper range of SBLOCAs) are sufficiently large to depressurize the RCS to approximately that of the containment pressure as shown in Figure 1. These breaks 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 uncovering and cladding heatup. For breaks with the RC pumps running for the first two minutes after LSCM, this situation is exacerbated, because the pumps push additional ECCS and RCS liquid to the break site. The duration of the uncovering 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 two to ten minutes post LOCA, therefore HPI inflow during these first several minutes is of little consequence for core cooling prior to the time of core refill so long as the LPI liquid reaches the vessel. After the CFTs are empty and the core is refilled, however, LPI and HPI flow provide both diversity of makeup injection sites and more than sufficient

ECCS flow to match the core boiloff rates. (NOTE: The dependency on HPI is greater in the event of a CFT line break as no LPI flow reaches the vessel. In this special break configuration, the unbroken CFT and HPI flow must be capable of providing for the necessary core cooling.)

Category 6: LBLOCAs

Break sizes greater than 0.75 ft^2 (up to a full double-ended break of any RCS pipe) are considered large break LOCAs. These break sizes are of sufficient size to cause the cladding to exceed the critical heat flux upon break initiation. If the break is on the cold leg side of the core, the 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. The pressure response for these breaks is shown in Figure 1.

Mode 3 Leaks or LOCAs at Low Decay Heat

The descriptions provided above are based on a LOCA occurring while the reactor is at full power with high core decay heat levels. For low decay heat Mode 3 considerations the RCS behavior is different. There is no forcing function for RCS repressurization because any break size relieves the core generated energy. Without substantial core heat generation, the secondary side has no potential to repressurize to the main steam safety valve pressures. (Note: If the RCPs are in operation, the RCPs will add sufficient energy to the coolant such that SG heat transfer will be required.) Therefore, the full power case descriptions do not necessarily represent the transient behavior when there is little to no core power; however, the break ranges are reasonable for dividing the spectrum into manageable sizes for discussion and evaluation for a low power Mode 3 LOCA.

Section 4 of the DBNPS Mode 3 Low Decay Heat Evaluation provides the discussion on the Mode 3 LOCA scenarios for the break classifications discussed above, although there was no real discussion on leaks. A leak is successfully managed with operator actions and makeup pump flow as described for the full power analyses. Its pressure response, shown in Figure 2 for the short-term period, is not unlike 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. The RCS pressure evolves to a pressure in which the ECCS injection matches the break flow. The pressure response will be similar with HPI only or when HPI and MU flow are credited. The only difference is if MU flow 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 steam generator secondary side pressure as shown in Figure 2. When there is no core energy addition, there is no need

for steam generator heat transfer, and 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. Their trends are also shown in Figure 2.

It should be noted that the RCS pressure trends shown in Figure 2 are based on considerations of minimal ECCS flow (i.e. one HPI pump and one LPI pump) for the LOCA categories. If additional ECCS is available for the leaks and Category 1 and 2, the RCS pressures will be slightly higher.

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

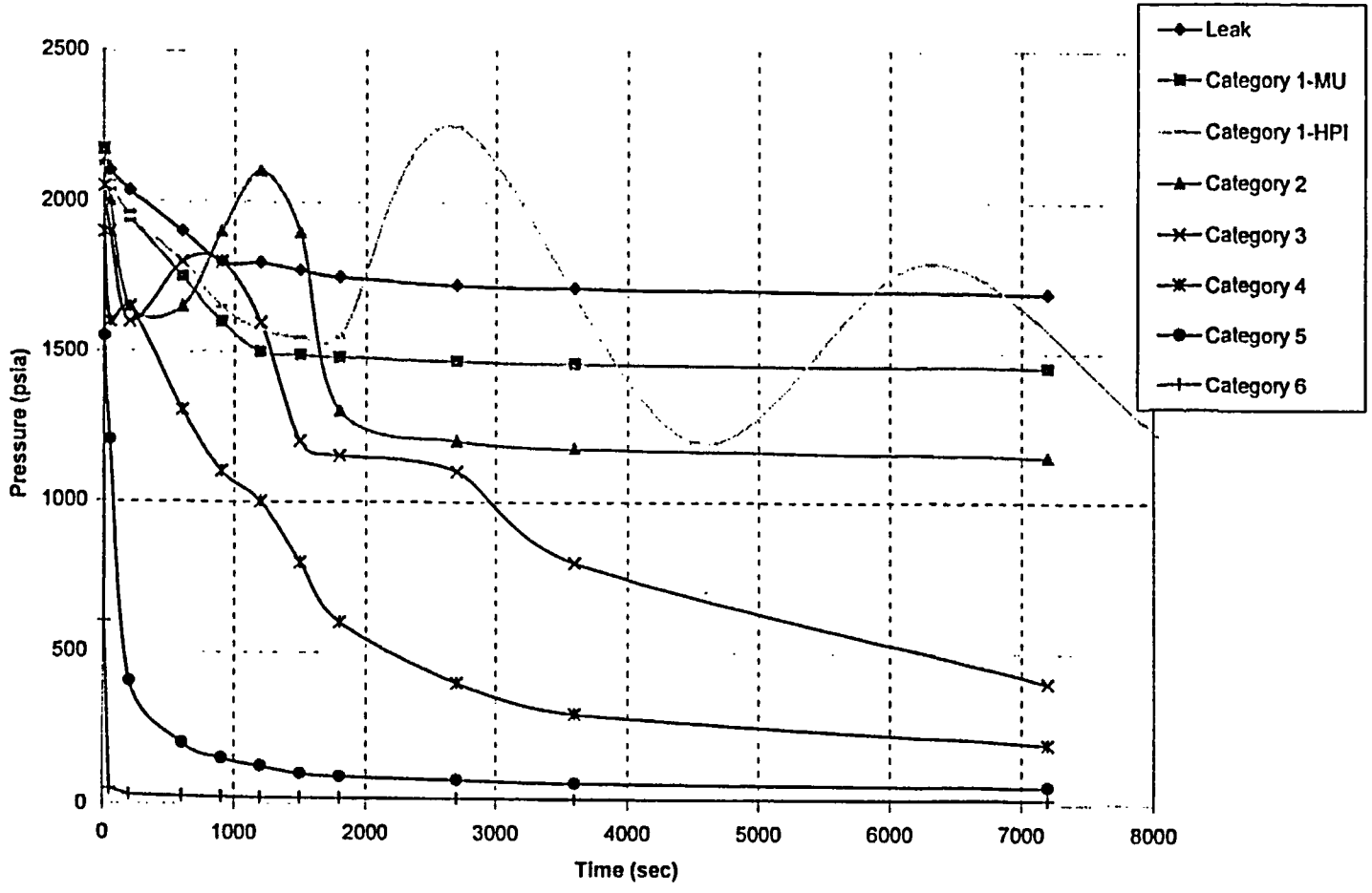
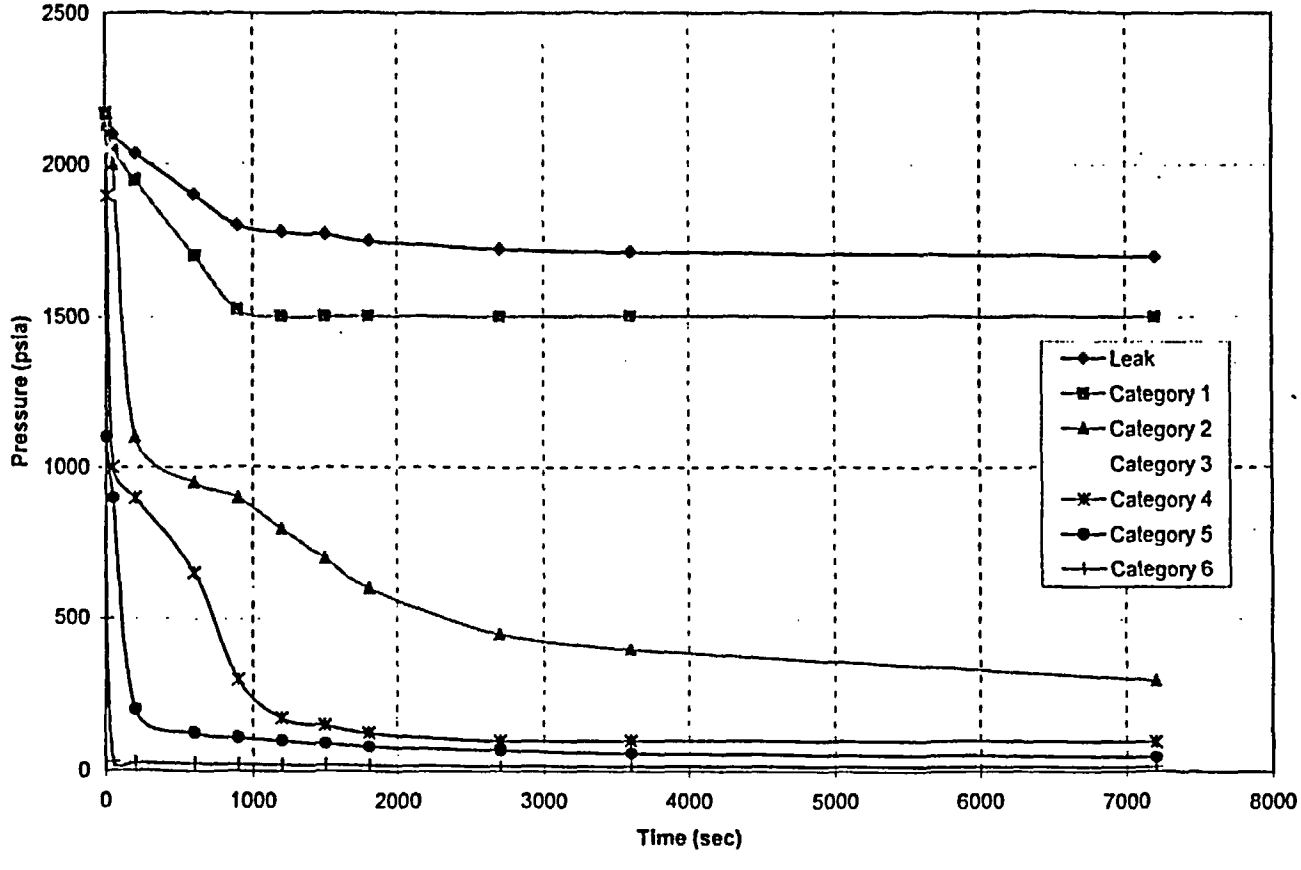


Figure 2. Mode 3 Low Decay Heat Post-LOCA Representative Pressure-Time Histories



Docket Number 50-346
License Number NPF-3
Serial Number 2980
Enclosure 3

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
To provide additional confidence in the quality of the water contained in the BWST, prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008, 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 Reactor Coolant System (RCS) as follows: 1. 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. 2. 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.	Prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008.

In addition to the BWST sampling discussed in the Question 1 response, prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008, if the HPI pumps are modified to incorporate internal strainers, the piping associated with HPI pump operation will be flushed as follows: 1. 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. 2. 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. In addition, the section of piping between the containment emergency sump and valves DH9A and DH9B will be inspected for debris prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008.	Prior to operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008.
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.
The HPI pumps will be available (i.e., capable of being placed in service within the time they are required) for use in a piggy-back configuration when ECCS is aligned for containment emergency sump recirculation as a defense-in-depth measure. 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.	During operation in Mode 4 and Mode 3 under the exception proposed by LAR 03-0008.

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 NOP test.	Prior to participating in operations associated with the Mode 3 NOP test.
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 NOP test.	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 NOP test.
Potentially affected Technical Support Center (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 NOP test.	Prior to participating in operations associated with the Mode 3 NOP test.
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 NOP test.	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 NOP test.
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.