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10 CFR 50.90
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License Number NPF-3

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Subject: Davis-Besse Nuclear Power Station
Exigent License Amendment Application to Modify Technical Specification 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$
(License Amendment Request No. 03-0008)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90 and 50.91, an exigent license amendment is requested for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed amendment would consist of a modification of Technical Specification (TS) 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$, applicable only during the ongoing Thirteenth Refueling Outage. Technical Specification Limiting Condition for Operation (LCO) 3.5.2 includes a requirement to have an operable Emergency Core Cooling System (ECCS) flow path capable of taking suction from the Borated Water Storage Tank (BWST) on a safety injection signal and manually transferring suction to the containment emergency sump during the recirculation phase of operation. This LCO is applicable in Operational Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Standby).

The proposed change to LCO 3.5.2 would add an exception applicable only during the Restart Test Plan inspection activities conducted during the ongoing Thirteenth Refueling Outage. The DBNPS reactor has been shut down since February 16, 2002, and 76 of 177 irradiated fuel assemblies have been replaced with fresh (unirradiated) fuel assemblies. Based on these

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circumstances, the current decay heat generation rate of the reactor core is a very small fraction (less than approximately 1%) of the decay heat normally assumed in accident analyses. In addition, since the reactor will be maintained subcritical during the Restart Test Plan activities, the decay heat generation rate will be essentially constant at that small value. Under these conditions, for the proposed Mode 3 Restart Test Plan inspection activities, evaluations have been performed that demonstrate that the acceptance criteria of 10 CFR 50.46 will be met, even if the HPI pumps are not capable of operating in the recirculation phase. This evaluation used conservative assumptions compared to the assumptions in the present accident analysis. Multiple Mode 3 entries may be necessary under the proposed exception if leakage is discovered in any part of the RCS and re-inspection is needed following corrective action. Operation in Mode 1 or Mode 2 while relying upon the provisions of this exception is prohibited. Enclosure 1 to this letter contains the technical reasoning for these proposed changes and the proposed no significant hazards consideration determination. Enclosure 2 provides an evaluation of low reactor core decay heat Mode 3 loss-of-coolant accidents (LOCAs) for the DBNPS.

The Restart Test Plan provides an opportunity to identify potential issues with safety-grade equipment as well as equipment important to plant reliability. A delay in completion of the Restart Test Plan would result in a delay in the correction of these currently unidentified equipment issues, with a potential corresponding delay in other station equipment repairs and inspections. The enclosed proposed license amendment application involves no significant increases in plant risk. Potential consequences have been reviewed and are bounded by the present accident analysis, which assumes a LOCA from full power operation with a limiting reactor core decay heat generation rate.

The ECCS is designed to mitigate the consequences of a LOCA, providing both short-term and long-term core cooling capability. In the initial "injection phase" of a LOCA, the High Pressure Injection (HPI) pumps and Low Pressure Injection (LPI) pumps provide ECCS injection into the Reactor Coolant System (RCS), drawing suction flow from the BWST. Prior to the exhaustion of the BWST inventory, the suction of the LPI pumps is manually transferred to the containment emergency sump, beginning the "recirculation phase." At this time, provided that the LPI flow rate exceeds a minimum requirement, HPI flow is terminated. If the RCS pressure is such that the LPI flow rate is below the minimum requirement, the HPI pumps are required for the recirculation phase. In this event, the LPI pumps are lined up as booster pumps for the HPI pumps in a "piggy-back" mode.

During the ongoing Thirteenth Refueling Outage, the FirstEnergy Nuclear Operating Company (FENOC) identified via the station's corrective action program that the HPI pumps, if required to operate in the "piggy-back" mode, are potentially susceptible to damage from debris small enough to pass through the openings in the containment emergency sump screens. Although operation of the HPI pumps in piggy-back mode could be impacted, this condition would not affect operation of the pumps during the earlier ECCS injection of water from the BWST. Pending resolution of this issue, both HPI trains have been declared inoperable. Due to its complexity, resolution of this issue may take several months, and may include replacement or modification of the existing HPI pumps.

While resolution of the HPI pump issue proceeds, preparations are underway for implementation of the Restart Test Plan, which incorporates many of the lessons-learned from other plants that have restarted from extended outages. The Restart Test Plan has been discussed with the NRC during several public meetings. The plan includes inspections of plant equipment that have undergone maintenance during the outage. The plan also includes several pressure inspections of the RCS. One focus of the Restart Test Plan is to confirm that there is no leakage from the incore monitoring instrumentation nozzles penetrating the reactor vessel bottom head. This inspection was discussed with the NRC staff during meetings on November 13, 2002, November 26, 2002, and April 4, 2003. The DBNPS, in conjunction with Framatome ANP, has concluded that incore monitoring instrumentation (IMI) nozzle leakage is highly unlikely. This conclusion is based on the absence of boron deposit build-up on the bottom head insulation, the absence of expected radionuclides in samples obtained from a number of IMI nozzles, and the relatively low concentration of lithium in the samples. A partial pressurization and inspection of the RCS in accordance with the Restart Test Plan is expected to confirm that there are no visible leaks prior to entering Mode 3.

The DBNPS is currently in Mode 5 (Cold Shutdown), with the RCS filled and pressurized to approximately 50 psig. The remaining outage activities are being completed in preparation for entering Mode 3, in support of the current schedule. As mentioned above, the Restart Test Plan will include several pressure inspections of the RCS, including a normal operating pressure and temperature inspection that will require entry into Mode 3 since the conditions would exceed 280 °F. These inspections will serve to identify RCS leakage for various RCS components, including the reactor vessel incore monitoring instrumentation nozzles. The normal operating pressure and temperature inspection involves a hold time of seven days at approximately 2155 psig and 532 °F. The heatup will be accomplished utilizing the minimal core decay heat currently present and the heat added by operation of the Reactor Coolant Pumps.

At this time, the FirstEnergy Nuclear Operating Company requests that this license amendment application be processed in the manner provided by 10 CFR 50.91(a)(6) for exigent circumstances. The present basis for the exigent circumstances is provided in Enclosure 1 to this letter. Currently, the DBNPS expects to enter Mode 3 within 30 days. FENOC requests issuance of the amendment to support this milestone. To support this request, FENOC is available to meet with the NRC staff. Should FENOC determine that a change to the Mode 3 entry date will affect the exigent circumstances of this application, FENOC management will promptly notify the NRC staff.

The proposed changes have been reviewed by the DBNPS Station Review Board and Company Nuclear Review Board.

Should you have any questions or require additional information, please contact Mr. Patrick J. McCloskey, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

Lew W Myers

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Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
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Utility Radiological Safety Board

APPLICATION FOR AMENDMENT

TO FACILITY OPERATING LICENSE NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This submittal requests changes to the Davis-Besse Nuclear Power Station Unit Number 1, Facility Operating License Number NPF-3. The statements contained in this submittal, including its associated enclosures and attachments, 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: 5/14/03

By: Lew W. Myers

Lew W. Myers, Chief Operating Officer

Docket Number 50-346
License Number NPF-3
Serial Number 2950
Enclosure 1

DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 03-0008

(36 pages follow)

**DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 03-0008**

Subject: Proposed Change to Limiting Condition for Operation (LCO) 3.5.2 to Add an Exception Regarding HPI Train Operability in Mode 3

1.0 DESCRIPTION

This letter is a request to amend the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3.

The proposed change would revise the Operating License to add a limited exception regarding high pressure injection (HPI) train operability to Limiting Condition for Operation (LCO) 3.5.2, "Emergency Core Cooling Systems – ECCS Subsystems – Tavg ≥ 280 °F," in Mode 3 only.

2.0 PROPOSED CHANGE

The proposed change would add an asterisked footnote to LCO 3.5.2 that would read as follows:

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

System Health Assurance review activities conducted by the FirstEnergy Nuclear Operating Company (FENOC) during the ongoing Thirteenth Refueling Outage identified via the station's corrective action program that the HPI pumps, if required to take suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a loss-of-coolant accident (LOCA), are potentially susceptible to damage from debris small enough to pass through the openings in the containment emergency sump screens. This condition would not affect operation of the pumps during the earlier injection of water from the Borated Water Storage Tank (BWST). Pending resolution of this issue, both HPI trains are currently declared inoperable since the plant licensing basis requires that the HPI pumps be capable of operating in the piggy-back mode during the recirculation phase when specific criteria are met.

The Davis-Besse Nuclear Power Station (DBNPS) is currently in Operational Mode 5 (Cold Shutdown), with the Reactor Coolant System (RCS) filled and pressurized to approximately 50 psig. Prior to Mode 3 entry, the RCS will be pressurized to approximately 250 psig. The RCS temperature will remain less than 200 °F during the 250 psig inspection (i.e., Mode 5). Visual inspections for RCS leakage will be performed on various RCS components, including the reactor vessel incore monitoring instrumentation (IMI) nozzles. Following successful completion of these inspections, the plant will be raised to normal operating pressure (approximately 2155 psig) and a temperature of approximately 532 °F and held there for approximately seven days to confirm that there is no RCS pressure boundary leakage. This heatup will be accomplished utilizing the minimal core decay heat currently present and the heat added by operation of the Reactor Coolant Pumps. The conditions exceed 280 °F and will therefore require entry into Mode 3. The 250 psig and 2155 psig inspections are described in the

"Davis-Besse Restart Test Plan." The Restart Test Plan has been discussed in the "Davis-Besse Return to Service Plan" previously submitted to the NRC.

Under the provisions of LCO 3.0.4, entry into an operational mode or other specified applicability condition shall not be made unless the conditions of the LCO are met without reliance on provisions contained in the Action statements unless otherwise excepted. This provision would preclude entry into Operational Mode 3 (Hot Standby) without two operable HPI trains. The proposed change to LCO 3.5.2 would add an exception applicable only during the Restart Test Plan and RCS leakage inspection activities conducted during the ongoing Thirteenth Refueling Outage. For entry into Mode 3, both HPI trains will be required to be operable except for the capability of taking suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a loss-of-coolant accident. Multiple Mode 3 entries may be necessary under the proposed exception if leakage in any part of the RCS is discovered and re-inspection is needed following corrective action. An exception to the above is that if IMI nozzle leakage is discovered, the proposed exception would not be utilized for a Mode 3 entry following corrective action. Operation in Mode 1 or Mode 2 while relying upon the provisions of this exception is prohibited.

3.0 BACKGROUND

The requirement to have an operable flow path capable of taking suction from the BWST on a safety injection signal and manually transferring suction to the containment emergency sump during the recirculation phase of operation is included in TS Limiting Condition for Operation (LCO) 3.5.2. This LCO is applicable in Operational Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Standby).

The ECCS equipment required by the Technical Specification is described in Section 6.3 of the DBNPS Updated Safety Analysis Report (USAR). The ECCS is designed to mitigate the consequences of all breaks of the RCS pressure boundary which result in loss of reactor coolant at a rate in excess of the capability of the Reactor Coolant Makeup System, up to and including a break equivalent in area to the double-ended rupture of the largest pipe of the RCS. The break spectrum also considers breaks in the HPI and Core Flood Tank lines. The analyses make bounding assumptions regarding initial reactor power, decay heat generation rate, and other operational considerations.

During ECCS operation, the suction flow for the HPI, LPI, and containment spray pumps is initially provided by the BWST. Prior to the exhaustion of the BWST inventory, LPI and containment spray suction flow is transferred to the containment emergency sump. Before this transition, the operator will determine, based on the LPI flow indicator, whether the LPI system needs to be cross-connected to the HPI system. If LPI flow is not above a predetermined rate, the operator is instructed to connect the LPI pumps to operate as booster pumps for the HPI pumps in a "piggy-back" mode. This connection is necessary to provide a flowpath and to ensure adequate net positive suction head (NPSH) for the HPI pumps to operate, and results in continued ECCS flow from the containment emergency sump when RCS pressure is above the discharge pressure of the LPI system.

The containment emergency sump is protected by screens designed to limit the passage of debris during a loss-of-coolant accident. The HPI pumps, if required to operate in the piggy-back mode during the recirculation phase, are potentially susceptible to damage from debris small enough to pass through the containment emergency sump screens. Pending resolution of this issue, both HPI trains have been declared inoperable. The current Technical Specification LCO 3.5.2 would preclude entry into Mode 3 without two operable HPI trains capable of meeting the accident analysis requirements assuming a LOCA from full power operation.

4.0 TECHNICAL ANALYSIS

4.1 Deterministic Evaluation

4.1.1 Conditions Unique to the Proposed Mode 3 RCS Leakage Inspection

For the proposed inspection, the enclosed evaluation demonstrates that installed safety-grade equipment is capable of cooling and depressurizing the plant to conditions where LPI can inject from the containment emergency sump prior to depletion of the BWST inventory. This evaluation used conservative assumptions compared to the assumptions in the present accident analysis.

The DBNPS reactor has been shut down since February 16, 2002, more than 14 months, and 76 of 177 fuel assemblies have not been irradiated. The current decay heat generation rate of the reactor core is a very small fraction (less than approximately 1%) of the decay heat used in the present accident analysis, and it would be essentially constant at that small value. The low decay heat essentially makes the dominant accidents less challenging by providing much greater margins in safety-related injection flows, greater margins in operator response times, greater margin in feedwater capacity, greater margins in heat exchanger capacities, longer heat-up times, and slower boiling rates.

A number of postulated accidents utilize natural circulation and primary-to-secondary heat transfer to remove decay heat. Since natural circulation is thermally driven, the low decay heat in the current reactor core will result in less driving head for natural circulation. Therefore, under conditions where natural circulation would normally be relatively vigorous, natural circulation might be intermittent. However, as long as the RCS hot legs remain filled, natural circulation would periodically resume as needed, based on the intermittent development of a sufficient primary to secondary temperature difference. In postulated LOCAs, where RCS inventory is reduced, natural circulation could be lost at an earlier time due to the lower driving head. If natural circulation is lost due to inventory loss, natural circulation would also be less likely to resume. However, with very low decay heat, any given break size will be more capable of reducing RCS pressure. This facilitates reaching a pressure where HPI is not required and injection via LPI will maintain core cooling without natural circulation. Additionally, safety grade and non-safety grade systems are available to assure that the RCS can be adequately depressurized.

Normally, postulated accidents are analyzed to occur from full power conditions with bounding decay heat. For a large break LOCA, the high level of stored energy in the fuel results in bounding peak cladding temperature excursions. For certain small-break LOCAs, such as core flood line breaks and HPI line breaks, the high decay heat also results in relatively large clad temperature excursions. In the Mode 3 evaluation performed for this license amendment application (summarized below), although different operator actions are required to be credited, either no clad temperature excursion is predicted to occur due to no core uncovering, or the excursion is bounded by the Mode 1 cases previously analyzed.

Additional benefits in risk are available compared to analyses conducted at full rated thermal power. At full power, a loss of offsite power is assumed. The most probable cause is that a reactor trip produces an upset to the electrical transmission grid and a fast bus transfer from the Auxiliary Transformer to the Startup Transformers. However, in the Mode 3 inspection, there would be no event induced grid disturbance due to a reactor trip and no transfer. Therefore, there is a high likelihood that offsite power will remain available. Section 5.2 describes additional compensatory measures that will be implemented during the period under which the proposed exception is effective, including limitations on activities in the plant's offsite power switchyard and electrical switchgear rooms. These compensatory measures will further reduce the risk of losing offsite power.

The current curie content in the reactor core is a small fraction of that assumed in the present accident analysis. This is discussed relative to the Maximum Hypothetical Accident (MHA) later in Section 4.1.4.

As described earlier in Section 2.0, prior to the Mode 3 inspection, Mode 5 leakage inspections at approximately 250 psig will be performed. These will include visual inspections of various components of the RCS, including the IMI nozzles. During Mode 3 conditions, relevant areas of containment will be accessible or monitored. This provides added assurance of the integrity of the RCS and minimizes the potential for leakage to progress into a break.

With respect to analysis, the HPI pumps will be credited to operate only with suction from the BWST. Since the BWST is a clean source of water, the operability of the existing HPI pumps is not challenged in this mode of operation. However, the DBNPS is licensed as a hot-standby plant, and normally relies on non-safety grade equipment to attain cold shutdown conditions.

4.1.2 Affected Accident Analysis

Relative to the proposed license amendment application, any accident analysis where the HPI system may be required to function in the piggy-back mode during the recirculation phase is of interest. A review of the DBNPS licensing basis shows that the only postulated accidents that require HPI in the piggy-back mode during the recirculation phase are particular break size ranges within the LOCA spectrum. Specifically, small-break LOCAs up to 0.1 ft^2 in size may be too small to effectively depressurize the RCS and, therefore, require evaluation. In addition, the special case of the Core Flood Line Break (0.44 ft^2) also requires evaluation.

The LOCA cases are discussed for Mode 3 conditions with the current reactor core decay heat generation rate below and in the enclosed Framatome ANP evaluation. It should be noted that some additional accidents, such as steam generator tube rupture and the "beyond design basis" loss of all feedwater accident, do not rely on HPI on containment emergency sump recirculation for design basis accident mitigation, but are of interest from a risk standpoint. These cases are discussed later in Section 4.2.

4.1.3 Mode 3 Loss-of-Coolant Accident (LOCA) Evaluation

An evaluation of the post-LOCA core cooling capabilities without credit for HPI flow during the recirculation phase, considering the extremely low reactor core decay heat generation rate under current station conditions, was performed by Framatome ANP. This analysis conservatively assumes no thermal losses from the RCS to the ambient containment atmosphere. The full report is provided as Enclosure 2 to the license amendment application submittal. As discussed in this report, a representative small-break LOCA analysis and related LOCA evaluations were completed by Framatome ANP and demonstrate adequate core cooling performance for the entire spectrum of required break sizes and locations. The evaluations concluded that the acceptance criteria of 10 CFR 50.46 will be met and that it is acceptable to perform the Mode 3 normal operating pressure inspection with the current HPI pumps. Potential consequences have been reviewed and are bounded by the present accident analysis, which assumes a LOCA from full power operation with a limiting reactor core decay heat generation rate.

In the unlikely event that a LOCA should occur during Mode 3 operation under the proposed exception, the ECCS systems will actuate to mitigate the consequences of the event. The HPI, Core Flood Tanks (CFT), and LPI systems work collectively (possibly with the non-safety makeup pumps when taking suction from the BWST) to supply ECCS flow to the RCS. The secondary side feedwater systems (both main feedwater and auxiliary feedwater (AFW) from the turbine driven AFW pumps) will provide the necessary flow to remove RCS energy not removed by the break. Because the decay heat generation rate is extremely low, the ECCS and AFW systems would have little difficulty keeping the core cool, provided the pump operation credited in the full power LOCA analyses is preserved. However, even with assuming that the HPI pumps are not available during the recirculation phase, and crediting only safety-grade equipment availability and considering any limiting single-failure, the evaluation shows that the postulated accidents can be mitigated safely with full compliance to the 10 CFR 50.46 acceptance criteria, and are bounded by the present accident analysis.

Under normal circumstances, the HPI pumps are relied upon during the post-LOCA long-term containment emergency sump recirculation phase to supply the following identified functions:

1. At least one HPI pump must supply ECCS flow when the RCS pressure is above the pressure at which LPI flow is delivered to the RCS.

2. At least one HPI pump must supply ECCS flow if only one LPI pump is providing ECCS flow and the LPI pump is not providing at least 1000 gpm of flow to each Core Flood Tank nozzle.

3. The HPI pump must supply flow to the auxiliary pressurizer spray line for use in core boron precipitation control if needed.

Under the present decay heat conditions, the LPI pumps can easily supply the required ECCS flow for long-term cooling without reliance on the HPI pumps.

<u>Expected LOCA progression for LOCAs in the size range between approximately 0.002 to 0.035 ft^2:</u>

Subcooling margin is not expected to be lost for the smallest breaks in this size range, and it is highly likely that offsite power will continue to be available. With adequate subcooling margin maintained and power available, the Reactor Coolant Pumps (RCPs) would be expected to remain in operation. The expected operator action for the smaller break sizes in this range is to depressurize the RCS by utilizing the normal secondary side equipment that was in service prior to the break, or at a minimum, to continue steam generator heat removal via the AFW Pump Turbine exhaust steam flow or atmospheric vent valves as required. Throttling of the ECCS injection from the HPI pumps to manage core exit subcooling margin would help reduce RCS pressure. This throttling is in accordance with existing procedural guidance. If a loss of offsite power were to occur, or for intermediate sized breaks in this range that would lose subcooling margin, the RCPs would not continue to run. For the smaller breaks, supplemental venting may be required as described below. For intermediate breaks, a cooldown utilizing the steam generator equipment should reach LPI injection pressure before the inventory in the BWST would be expended if procedures are followed. The larger sized breaks in this range, as well as breaks larger than this range, would depressurize by themselves, to where LPI would be effective, before the BWST inventory would be expended.

In the event that the depressurization is not accomplished by steam generator (SG) heat transfer or by the break itself, the pilot-operated relief valve (PORV) would be opened to reduce RCS pressure. Although the PORV solenoid and controls are not currently safety-grade, it is powered from essential DC power, and was refurbished in the Thirteenth Refueling Outage. Thus, it is expected to be highly reliable. The PORV is in the process of being upgraded to safety-grade. Upon completion of the necessary environmental qualification packages, the PORV will be considered safety-grade. This activity is expected to be completed prior to Mode 1 (Power Operation). If the PORV is opened, its effective flow area is sufficient to reduce RCS pressure to where LPI can inject prior to expending the BWST. In the event that the PORV is not available, other RCS vent paths, such as the RCS hot leg and pressurizer vents, may be successful in depressurizing the RCS depending on break size and location. However, these vent paths would not be effective for all potential break sizes and locations. This progression relies primarily on non-safety-grade equipment.

In the event that a cooldown using both secondary equipment and the PORV failed to accomplish the desired depressurization, ECCS throttling with the RCS not adequately subcooled would be required in order to reduce RCS pressure to where LPI can inject, prior to expending the BWST inventory. When the HPI flow is throttled, the ECCS injection flow will not match the flow out the break, resulting in a net RCS inventory loss that will cause the RCS to depressurize. As the RCS depressurizes, the saturation pressure for the core exit temperature will be reached, and RCS liquid will begin to flash to steam. Steam bubbles will collect in the upper head of the reactor vessel and in the high points of the hot legs. Throttling of ECCS flow with inadequate subcooling margin is not consistent with the current emergency operating procedure, and therefore, provisions are required, as described in the "Summary of Additional Operator Action Requirements for the Mode 3 Normal Operating Pressure RCS Leakage Inspection" (below) to address this situation. The operators would have the RCS hot leg level monitoring instrumentation available to monitor RCS liquid inventory, provided the plant computer was available. This instrumentation would be available until RCS level dropped below its range. The inventory of liquid in the core could then be indirectly monitored via the core exit thermocouples (the DBNPS does not have an installed reactor vessel level instrumentation system). The operators would be able to reestablish core cooling by increasing HPI flow.

As discussed in Section 3.2.2 of the enclosed Framatome ANP evaluation, an analysis of a conservative rapid HPI throttling ramp with the RCS not adequately subcooled indicates that the coolant level would lower to approximately one foot above the core before LPI injection would begin to refill the level. The evaluation indicates that even if the core is assumed to uncover, and even with no credited heat transfer, it would require over one hour for the core hot spot to reach 1400 °F. This would allow time for operators to reinitiate cooling by increasing HPI flow as described above. As the event progresses, the sensible heat stored in the RCS metal structures will dissipate, RCS pressure will decrease, and LPI flow to the core will commence.

With the operator action to decrease the RCS pressure to the LPI flow range, there is sufficient ECCS flow such that the core cooling requirements are met. However, the generic emergency operating procedures specify that HPI cannot be terminated until at least 1000 gpm of LPI flow can be delivered to each CFT nozzle. Achieving at least 2000 gpm of total LPI flow is not possible for these break sizes. Therefore, the HPI termination criterion will not be met. Given that the decay heat is so low for the planned Mode 3 RCS leakage inspection, it is reasonable to reduce the LPI flow requirement to 500 gpm per line before HPI is terminated. The basis for decreasing the LPI flow requirement is based on the minimal core boiloff rate for the Mode 3 inspection. The operator action to terminate HPI under the 500 gpm criterion will be the same as the operator action under the 1000 gpm criterion based on the present analysis.

It is also important to note that very small LOCAs will produce smaller amounts of containment debris. These break sizes will also require very low ECCS flow rates during the sump recirculation phase. The containment spray actuation will not occur for these break sizes, therefore the low ECCS flow rates from containment will transport the least amount of foreign material to the containment emergency sump. Therefore, continued HPI pump operation during containment emergency sump recirculation would be an option available to the operators under these scenarios.

Expected LOCA progression for Core Flood Tank (CFT) Line Break LOCAs (0.44 ft^2):

As described below, the evaluation shows that alternate flow paths for mitigation of a CFT line break are adequate, considering the low decay heat conditions that would exist during the Mode 3 RCS leakage inspection. With the exceptionally low decay heat generation rate, over two hours would be available to reestablish ECCS flow.

A complete severance of the CFT line will rapidly cause RCS pressure to decrease to the LPI injection pressure, but in this specific case, HPI is still initially required and available. Each of the two CFT lines shares a reactor vessel injection nozzle with the associated LPI train. Therefore, for a CFT line break, assuming a single-failure of the opposite side electrical bus, no injection from LPI is initially credited. In this case, the available LPI flow discharges out the break and does not reach the core. The de-energized LPI train is idle. Therefore, the intact CFT and one HPI train are credited for ECCS injection for this case. If this break occurs, the HPI system will function as designed, and the running HPI pump will replenish the reactor level prior to expending the BWST inventory. Thereafter, the LPI cross-tie needs to be opened and flow to the reactor vessel balanced between the two lines near the time the BWST is emptied. This allows

LPI flow to reach the core, eliminating the need for HPI during containment emergency sump recirculation. Opening the LPI cross-tie is an operator action that is already procedurally in place to address abundant cooling, but has not been previously credited in the LOCA analyses. This action is similar to the action in the present analysis of balancing HPI flow.

If the LPI flow path cannot be established via the cross-tie, or flow indication is not available, then LPI flow through the auxiliary pressurizer spray (APS) line is an available redundant flow path.

Boron Precipitation Control (BPC) for Long Term Cooling:

It is also necessary to maintain the capability to provide for long-term post-LOCA boron precipitation control, since the primary method relies on HPI Pump 1-2, as described in further detail below.

The current BPC methods are described in USAR Section 6.3.3.1.2.1, "Boron Precipitation Control." The plant design includes two active means of ensuring the chemical additive concentration remains below its solubility limit throughout the post-accident cooling period. The primary method of BPC utilizes HPI Pump 1-2, in piggy-back with LPI Pump 1-2, to supply water to the auxiliary pressurizer spray APS line via a tie-line, providing dilution flow to the pressurizer. This primary method is affected by the potential inability of the HPI pump to draw suction from the containment emergency sump (via the LPI pump) during the planned Mode 3 RCS leakage inspection, as discussed in further detail below. The backup BPC method utilizes one of the two operating LPI pumps taking suction from the decay heat removal (DHR) drop line and discharges a low (throttled) flow rate into the reactor vessel via the core flood nozzles. The backup method is unaffected by the proposed license amendment application.

BPC is not required when the RCS is above 322 °F, as insufficient boric acid is available in the entire RCS and ECCS to reach the solubility limit. BPC is also not required when the RCS below 322 °F and the core exit temperature is adequately subcooled, as no boiling occurs in the core region which could concentrate the boric acid.

Performance of the planned Mode 3 RCS leakage inspection is acceptable without a complete analysis of the post-LOCA BPC because the DBNPS has been shut down without any power generation since February 16, 2002. Since that time, 76 fresh (unirradiated) fuel assemblies have been loaded with 101 previously irradiated assemblies. The core decay heat is quite low, and will remain so during the planned Mode 3 RCS leakage inspection. 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.

Although it is not expected that BPC will be necessary, two BPC methods are available. For the planned Mode 3 RCS leakage inspection, if the core has lost subcooling margin and is at a pressure that typically requires BPC (i.e. less than 92 psia), sufficient time (greater than 6 days) exists to establish an appropriate boron dilution mechanism if one is deemed necessary. In the event that the primary BPC method is not available as a result of HPI Pump 1-2 inoperability, the LPI pump can provide adequate flow via the APS line to control the post-LOCA core boron concentration, considering the low decay heat level. It is estimated that this method would provide at least 20 gpm at 100 psi; 10 gpm would adequately remove the decay heat, with an additional 10 gpm to provide successful post-LOCA boron dilution.

<u>Summary of Additional Operator Action Requirements for the Mode 3 Normal Operating Pressure RCS Leakage Inspection:</u>

Depending on the postulated break size, assumed single-failure, and plant response, additional operator actions include the following, for smaller break LOCAs, a CFT line break, and long-term boron precipitation control, respectively:

1. For smaller break sizes, depressurization of the RCS to LPI pressure before BWST switchover will be performed. In some cases, this could involve throttling of HPI without maintaining subcooling margin.

2. Open the LPI cross-tie line to provide flow to both CFT lines, or initiate LPI flow through the auxiliary pressurizer spray line.

3. Initiate post-LOCA reactor core boron precipitation control as necessary with LPI flow through the auxiliary pressurizer spray line.

These actions are reasonable based on the time expected to be available to diagnose the situation and take the appropriate action. These operator actions will be described in the appropriate plant procedures prior to Mode 3 operation under the proposed license amendment exception. In addition, licensed operators will be trained to these new operator actions prior to standing shift in Mode 3 under the exception proposed by this license amendment application.

4.1.4 <u>Maximum Hypothetical Accident (MHA)</u>

For accidents occurring from full rated power, USAR Section 15.4.6.4, "Maximum Hypothetical Accident," demonstrates in a conservative manner that the operation of the station does not present any undue hazard to the general public. A hypothetical accident involving a gross release of fission products is evaluated. No mechanism is postulated whereby such a release occurs, since this would require a multitude of failures in the engineered safety features, which are provided to prevent such an occurrence. Fission products are assumed to be released from the core as stated in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," namely, 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids. For the planned Mode 3 RCS leakage inspection, since fuel pin pressure and cladding temperature excursions are bounded by a full power LOCA, the MHA release fractions are especially conservative.

The dose values under assumed full power accident conditions are shown in USAR Table 15.4.6-2, and are less than 10 CFR 100 limits. However, for the planned Mode 3 RCS leakage inspection, 76 of the fuel assemblies are freshly loaded new fuel assemblies that have no fission product isotopes. The 101 remaining fuel assemblies have had over 14 months of decay time. For comparison purposes, the curie contents of isotopes that could affect offsite dose were reviewed. Effectively, all radioactive isotopes of Xenon, Krypton, and Iodine (with the exception of Krypton -85) considered in the MHA will have decayed to zero curies. Thus, the upper bound on dose consequence for the proposed license amendment is a small fraction of the licensed MHA under the present conditions.

4.2 <u>Risk Evaluation</u>

Although this license amendment application is not "risk-informed," the following risk insights are provided.

At present, there are well-developed Probabilistic Safety Assessment (PSA) models for the DBNPS that evaluate accidents initiated from nominal full-power conditions and at shutdown. However, none of the models is directly suited to considering the unique conditions associated with pressurizing the RCS when the reactor core decay heat generation rate is very low. The models for the full-power PSA reflect the potential use of the HPI pumps in a containment emergency sump recirculation mode with respect to a loss of feedwater, steam generator tube rupture, boron precipitation control, and small-break LOCAs. The applicability of each of these HPI containment emergency sump recirculation models was reviewed considering the conditions expected to exist while in Mode 3 for the RCS leakage inspection.

A total loss of feedwater in Mode 1 leads to the need to initiate makeup/HPI cooling. If feedwater cannot be restored, the BWST inventory will eventually be depleted. The operators would open the PORV and establish HPI flow from sump recirculation, since the makeup pumps cannot be used in this mode. During Mode 3 operation under the proposed exception, if heat transfer via steam generators could not be restored, the operators would open the PORV to initiate feed and bleed cooling. With the PORV open, the decay heat load would be sufficiently low such that HPI flow from sump recirculation would not be expected to be required. If the PORV could not be opened, the RCS could not be depressurized, and feed and bleed cooling could not be established. Therefore, the ability of the HPI pumps to take suction from the containment emergency sump (via the LPI pumps) would not be relevant. For loss of feedwater sequences, any risk impact based on the availability of HPI flow from sump recirculation would be negligible.

Following a steam generator tube rupture (SGTR) from Mode 1 conditions, the RCS is cooled down to terminate the leakage through the broken tube. If a faulted steam generator is the only steam generator available for cooldown or if feedwater is not available, the operators will have to continue the cooldown using makeup/HPI cooling. If the BWST inventory is not replenished, the transfer to the emergency sump may be required before long term cooling is established. This case is essentially the same as a loss of feedwater discussed above and the risk impact based upon HPI pump recirculation availability in Mode 3 would be negligible.

The HPI pump is the primary post-LOCA boron dilution path with flow supplied to the auxiliary pressurizer spray line. Based on the enclosed Framatome ANP evaluation, the LPI pump should provide adequate flow to control the post-LOCA core boron concentration. Additionally, the time to initiate boron precipitation control provides a very long time window for initiation. Therefore, any change in risk from boron precipitation control at low decay heat generation rates would be expected to be negligible.

The CFT line break is explicitly modeled in the full power PSA. The PSA models the operator actions required to cross-connect LPI since these actions are proceduralized in the current operating procedures. With no credit for the HPI pumps following the injection phase, the contribution to core damage frequency (CDF) from the CFT line break is about 1.0E-9/year. This is based on a Medium LOCA frequency of 4.5E-5/year, a conditional probability of .02 that the break occurs in the CFT line, and a human error probability of about 1E-3 the operators fail to cross-connect before the BWST is depleted. Although not credited in the present USAR analysis, these operator actions are discussed in USAR Section 6.3.2.11, "Reliability Considerations."

Certain small-break LOCAs (SBLOCAs) result in depletion of the inventory in the BWST, and the RCS cannot be cooled down sufficiently to enter into normal shutdown cooling using the Decay Heat Removal (DHR) system, or to allow the LPI pumps to supply water from the containment emergency sump. If the steam generators and RCPs are available, steam generator depressurization would normally be expected to depressurize the RCS to the LPI pressure range. However, for RCP seal LOCA sequences the RCP would not be available and additional actions would be required to depressurize the RCS as described in the enclosed Framatome ANP evaluation. Although the risk impact of these additional actions is small due to the available time, a bounding estimation of the CDF was performed.

To bound the CDF impact for small LOCAs initiating event frequencies were obtained from NUREG/CR-5750, "Rates of Initiating Event at U.S. Nuclear Power Plants 1987-1995."

Small LOCA	5.0E-4 / yr
RCP Seal LOCA	2.5E-3 / yr
Total	3.0E-3 / yr

Due to the number of available means to depressurize the RCS to within LPI discharge pressure, hardware failure probabilities are significantly lower than human error probability for failure to depressurize. The operator action timing is dependent on the depletion of the BWST inventory, which provides a minimum time window of several hours. Therefore, a human error probability (HEP) of 0.01 would be reasonable. This probability is conservative based on the value of other HEPs calculated for the DBNPS PSA that are also based on the time to deplete the BWST inventory. Applying this HEP to the total initiating event frequency the core damage probability for a 200-hour period in Mode 3 can be calculated, as follows. The 200-hour Mode 3 period accounts for time to heat up and pressurize to normal operating temperature and pressure, a 7-day hold time for the RCS leak inspection, and time to cooldown and depressurize to Mode 4.

Core Damage Probability = 3.0 E-3 / yr (0.01) (200 hr / 8760 hr/yr) = 6.9E-07

A review was also made to determine if there were any other conditions requiring HPI flow from sump recirculation, not previously considered in the full-power PSA, which might be of interest under low power conditions. No additional sequences were identified.

In conclusion, the risk presented by entry into Mode 3 without the capability of the HPI pumps to take suction from the containment emergency sump (via the LPI pumps) is difficult to quantify with the available PSA model. However, based on a review of the existing PSA sequences the incremental CDF would be expected to be very small. A simplistic bounding assessment of the core damage probability for entry into this condition for 200 hours is 6.9E-7. Based on the guidance in EPRI TR-105396, "PSA Applications Guide," August 1995, this would be considered non-risk significant.

4.3 <u>Summary</u>

LOCA evaluations have been performed considering the current reactor core decay heat generation rate. These evaluations show that in the unlikely event a LOCA did occur while operating in Mode 3 under the proposed exception, the accident can be mitigated without crediting HPI flow during the recirculation phase, while crediting new operator actions similar to those assumed in the present USAR for balancing HPI flow. In addition, a risk evaluation has been performed and shows that the increase in core damage frequency, accounting for human error probability for additional operator actions, is very small.

The requested amendment provides for the adequate protection of the public health and safety. Specifically:

- The current HPI pumps are fully capable of performing their primary safety function of injecting water from the Borated Water Storage Tank into the reactor core.

- The HPI pumps could potentially be impacted by debris only while operating in the recirculation mode. The HPI pumps are needed in the recirculation mode only for certain small-break LOCAs in which the BWST has been depleted and the pressure of the RCS is above the injection pressure of the LPI system. However, it is unlikely that there would be any need to use the HPI pumps during recirculation for a small-break LOCA in Mode 3. Under existing plant procedures, there are numerous methods using non-safety-related plant systems for reducing the pressure of the RCS to the pressure of the LPI system prior to depletion of the BWST. In this event, there would be no need to use the HPI pumps in the recirculation mode.

- Even if it were assumed that all of the normal depressurization methods were to fail, there would be no adverse impact on safety. In particular, prior to depletion of the BWST, the operators could reduce the pressure of the RCS to the pressure of the LPI system by throttling the HPI pumps in the injection mode and allowing the break to reduce the pressure of the RCS. In this event, the LPI pumps (with proper interconnection of the LPI trains) could supply sufficient cooling water to the core.

- Even if there were operator error in throttling the HPI pumps or interconnecting the LPI trains, there would be sufficient time for the operators to recover and restore coolant flow to the core. Due to substantial radioactive decay in the 101 irradiated fuel assemblies and the replacement of 76 spent fuel assemblies with new fuel assemblies, the decay heat load in the reactor core is a very small fraction of normal decay heat loads. Given these low decay heat loads, it was conservatively calculated that the core could be uncovered for more than one hour without reaching 1400 °F (the threshold for significant metal water reaction).

- Even if there were to be a severe accident in Mode 3, the consequences to the public would not be significant. Due to substantial radioactive decay in the 101 irradiated fuel assemblies and the replacement of 76 spent fuel assemblies with new fuel assemblies, the source term for a severe accident involving the current core would be a small fraction of the source term provided in TID-14844 for a core that was operating at full power at the time of the accident. Thus, the consequences of an accident in Mode 3 would be substantially less than the consequences currently estimated in the DBNPS Updated Safety Analysis Report.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

The Emergency Core Cooling System (ECCS) is designed to mitigate the consequences of all breaks of the Reactor Coolant System (RCS) pressure boundary which result in loss of reactor coolant at a rate in excess of the capability of the Reactor Coolant Makeup System, up to and including a break equivalent in area to the double-ended rupture of the largest pipe of the RCS. During ECCS operation, the suction flow for the high pressure injection (HPI), Low Pressure Injection (LPI), and Containment Spray pumps is initially provided by the Borated Water Storage Tank (BWST) inventory. Prior to the exhaustion of the BWST inventory, LPI and Containment Spray suction flow is transferred to the containment emergency sump. This transition constitutes the beginning of the "recirculation phase" of the accident. Before this transition, the operator will determine, based on the LPI flow indicator, whether the LPI and HPI systems need to be cross-connected (piggy-backed). If LPI flow is not above a predetermined rate, the operator is instructed to connect the LPI pumps to operate as booster pumps for the HPI pumps. This connection is necessary to ensure adequate net positive suction head (NPSH) for the HPI pumps to operate from the containment emergency sump, and results in continued ECCS flow from the containment emergency sump even though RCS pressure may be above the discharge pressure of the LPI pumps.

It has been determined that the existing HPI pumps, if required to take suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a loss-of-coolant accident (LOCA), are potentially susceptible to damage from debris small enough to pass through the openings in the containment emergency sump screens. This condition would not affect operation of the pumps during injection of water from the BWST. Pending resolution of this issue, both high pressure injection trains are currently declared inoperable.

The proposed change to Technical Specification Limiting Condition for Operation (LCO) 3.5.2 would add an exception applicable only during Restart Test Plan and RCS leakage inspection activities conducted during the ongoing Thirteenth Refueling Outage. For entry into Mode 3, both HPI trains would be required to be operable except for the capability of taking suction from the containment emergency sump (via the LPI pumps) during the recirculation phase. Operation in Mode 1 or Mode 2 while relying upon the provisions of this exception would be prohibited.

A representative small-break LOCA analysis and related LOCA evaluations were completed by Framatome ANP without credit for HPI flow during the recirculation phase, considering the extremely low reactor core decay heat generation rate under current station conditions. These evaluations demonstrate adequate core cooling performance for the entire spectrum of required break sizes and locations. The evaluations used NRC-approved evaluation model analyses as necessary to validate any break scenarios that cannot be readily shown to be less limiting than a similar LOCA break size from the full power analyses. The evaluations concluded that the acceptance criteria of 10 CFR 50.46 will be met and that it is acceptable to perform the Mode 3 normal operating pressure inspection with the current HPI pumps.

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment." As discussed below, the proposed amendment does not involve a significant hazards consideration under the three criteria contained in Section 50.92(c):

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change allows the plant to operate in Mode 3 in support of RCS leakage inspection activities conducted during the ongoing Thirteenth Refueling Outage, utilizing a limited exception to Limiting Condition for Operation (LCO) 3.5.2. This LCO applies in plant operational Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Standby). Under the proposed exception, for entry into Mode 3, both HPI trains would be required to be operable except for the capability of maintaining suction from the containment emergency sump during the recirculation phase.

   The ability of the HPI pumps to draw suction from the containment emergency sump (via the LPI pumps) is a design feature credited by the Davis-Besse Nuclear Power Station Updated Safety Analysis Report (USAR) for mitigation of various types of loss-of-coolant accidents (LOCAs). Due to the potential susceptibility to damage from debris contained in the pumped fluid, the existing HPI pumps may not be capable of maintaining suction from the containment emergency sump without an increased probability for malfunction. However, the current plant conditions are unique in that the decay heat generation rate in the reactor core is extremely low due to the fact that the plant has not operated in more than 14 months and 76 unirradiated fuel assemblies have been loaded into the core, replacing irradiated fuel assemblies.

A LOCA evaluation has been performed considering the current reactor core decay heat generation rate. The evaluation shows that in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, the accident can be mitigated without crediting HPI flow during the recirculation phase, while crediting additional operator actions not presently credited in the USAR. In addition, a risk evaluation has been performed and shows that the increase in core damage frequency, accounting for human error probability for the additional operator actions, is very small. Also, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, radiological consequences would be very small compared to the accident analyses results of record, given the fission product decay over the extended plant shutdown. Therefore the proposed change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   There are no new or different accident initiators introduced by the proposed change to allow the plant to operate in Mode 3 under a limited exception, with the HPI pumps not capable of maintaining suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a LOCA. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change allows the plant to operate in Mode 3 under a limited exception, with the HPI pumps not capable of maintaining suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a LOCA. Although the ability of the HPI pumps to draw suction from the containment emergency sump (via the LPI pumps) is a design feature credited by the Davis-Besse Nuclear Power Station USAR for mitigation of various types of LOCAs, an evaluation shows that given the extremely low decay heat generation rate in the reactor core under current plant conditions, and crediting additional operator actions, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, the accident can be mitigated without crediting HPI flow during the recirculation phase. In addition, a risk evaluation has been performed and shows that the increase in core damage frequency, accounting for human error probability for the additional operator actions, would be expected to be very small. Also, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, radiological consequences would be very small compared to the accident analyses results of record, given the fission product decay over the extended plant shutdown. Accordingly, given that accident severity or consequences will not be significantly increased under the proposed change, a significant reduction in a margin of safety is not involved.

Based on the above, it is concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

As stated in the DBNPS Updated Safety Analysis Report (USAR), Appendix 3D, "Conformance with the NRC General Design Criteria, Safety Guides, and Information Guides," the design of the Davis-Besse Nuclear Power Station meets the intent of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," as published in the Federal Register on February 20, 1971, and as amended in the Federal Register on July 7, 1971.

Regarding General Design Criterion (GDC) 35, "Emergency Core Cooling," USAR Appendix 3D states:

> A system to provide abundant emergency core cooling is provided. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities are provided to assure that for onsite electric power system operation (assuming that offsite power is not available) and for offsite electric power system operation (assuming that onsite power is not available) the system safety function can be accomplished, assuming a single-failure.
>
> Abundant emergency core cooling is provided by the low-pressure injection (decay heat removal), high-pressure injection, and the core flooding systems. These three systems make up an Emergency Core Cooling System (ECCS) that maintains core cooling in the event of a loss-of-coolant accident (LOCA).

Redundancy of components, power supplies, and initiation logic and separation of functions are provided so that a single-failure does not prevent the ECCS from fulfilling its function. The ECCS may be operated from either onsite or offsite power supplies.

The primary function of the Emergency Core Cooling System is to deliver cooling water to the reactor core in the event of a LOCA. The system provides protection for all potential break sizes in the reactor coolant system pressure boundary piping up to and including the double-ended rupture of the largest pipe. In addition, breaks in the High Pressure Injection line and the Core Flood Tank line are postulated.

The basic design criteria for loss-of-coolant accident evaluation are as follows:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

b. The calculated total oxidation of the fuel cladding shall not exceed 17% of the total cladding before oxidation.

c. The amount of hydrogen generated from cladding metal-water reaction does not exceed 1% of the total amount of cladding in the reactor.

d. The core geometry is maintained in a state that is amenable to cooling.

e. The cladding temperature is reduced and maintained at an acceptably low value and decay heat is removed for extended periods of time.

For a rupture in the steam piping, the Emergency Core Cooling System adds shutdown reactivity, so that with minimum tripped rod worth and minimum ECCS operation, the reactor core does not return to criticality; thus, there is no core damage.

Reference: USAR Chapter 6 and 10 CFR 50.46

Pursuant to 10 CFR Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," paragraph (a)(1)(ii), the Davis-Besse Nuclear Power Station Emergency Core Cooling System (ECCS) is modeled in conformance with the required and acceptable features of 10 CFR 50, Appendix K, "ECCS Evaluation Models." Compliance with the 10 CFR 50.46 acceptance criteria is described in USAR Section 6.3, "Emergency Core Cooling System."

Based on the technical analysis (Section 4.0), under the current plant conditions (low decay heat generation rate), the intent of 10 CFR 50, Appendix A, GDC 35, and the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K will continue to be met for plant operation in Mode 3 (Hot Standby). The Framatome ANP Mode 3 LOCA evaluation (Enclosure 2 to this license amendment application submittal) demonstrates that if a LOCA occurs in Mode 3 with the current low decay heat generation rate, adequate core cooling will be maintained and, therefore, the acceptance criteria of 10 CFR 50.46 will be met.

As mentioned in Section 4.1.3 above, prior to Mode 3 operation under the proposed license amendment exception, the required operator actions to mitigate a LOCA will be described in the appropriate plant procedures. Licensed operators will be trained to the new operator actions prior to standing shift in Mode 3 under the proposed exception. In addition, during the period under which the proposed exception is effective, the following administrative controls will be implemented as additional compensatory measures:

1. The DBNPS will maximize the availability of plant 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). Required surveillance testing will continue to be performed, however, maintenance activities that adversely affect operability will not be performed. The systems and components include: Low Pressure Injection, Decay Heat Removal, Emergency Diesel Generators, Auxiliary Feedwater, 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. If any of these systems become inoperable for reasons other than performance of surveillance testing, or if the High Pressure Injection system becomes inoperable for reasons other than the capability of maintaining suction from the containment emergency sump (via the LPI system), the plant will initiate a cooldown within two hours to at least Mode 4, at a cooldown rate prudent for the plant conditions.

2. The DBNPS will limit activities in the plant's offsite power switchyard and electrical switchgear rooms to those of an essential nature.

3. An additional dedicated licensed operator (above the minimum TS manning requirement) will be on-shift in the control room to assist with the added operator actions which may be necessary to mitigate a Mode 3 LOCA without reliance on the HPI pumps drawing suction from the containment emergency sump (via the LPI pumps).

4. Core reactivity will be controlled in a safe shutdown condition by ensuring that the boron concentration is greater than or equal to the estimated "all rods out" critical boron concentration. Either the control rod groups will be fully inserted with trip breakers open, or a single safety group of control rods will be withdrawn to provide a prompt source of trippable negative reactivity. In addition, potential sources of boron reduction will be strictly controlled to ensure an inadvertent boron reduction does not take place (e.g. demineralized water sources, non-boron saturated purification demineralizers). Boric acid concentration will be sampled at least once per 12 hour shift. Source range nuclear instrumentation count rate will be logged at least once per 4 hours.

5. Operating personnel will be notified by Standing Order to ensure that the above actions are maintained.

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

6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 EXIGENT CIRCUMSTANCES

Under 10 CFR 50.91(a)(6), the NRC will designate an amendment as exigent if "a licensee and the Commission must act quickly and that time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment, and it also determines that the amendment involves no significant hazards considerations." The licensee must explain the exigency, why the licensee could not have avoided it, and why the licensee has used its best efforts to make a timely application for the amendment.

This license amendment application satisfies the criteria for treatment as an exigent amendment under Section 50.91(a)(6) as follows:

- No Significant Hazards Consideration – As discussed in Section 5.1 above, this amendment involves no significant hazards consideration.

- Exigent Circumstances – The DBNPS is currently in Mode 5 and expects to enter Mode 3 within 30 days. Given the time needed to publish a Federal Register notice of the amendment request, provide 30 days for public comment, evaluate any public comments, and issue the amendment, the NRC would not be able to issue the amendment until well after DBNPS is currently expected to be ready to enter Mode 3. Entry into Mode 3 is needed to support the performance of the Restart Test Plan, including the RCS leakage inspections that are critical path to restart of the plant. Therefore, the license amendment is needed to prevent a delay in the restart of the plant. Additionally, early performance of the RCS leakage inspections will, in turn, allow any potential RCS leakage repairs to be conducted in parallel with ongoing resolution of the issue with the HPI trains. Performing the RCS leakage repair (if required) and the HPI pump issue resolution activities in-parallel in lieu of in-series is expected to result in a schedule improvement for return to power from the ongoing Thirteenth Refueling Outage. Due to its complexity, resolution of the HPI pump issue may take several months and may include replacement of the existing HPI pumps. Should FENOC determine that a change to the Mode 3 entry date will affect the exigent circumstances of this application, FENOC management will promptly notify the NRC staff.

- Unavoidable Circumstances – The condition involving the HPI pumps is a design deficiency. To the best of FENOC's knowledge, there are no other operating plants with this type of HPI pump. The condition is not common and was not readily discoverable. The condition was only identified as a result of FENOC's aggressive efforts to verify the design of its systems. Therefore, FENOC could not reasonably have avoided this situation.

- Best Efforts for Timely Application – FENOC self-identified the susceptibility of the HPI pumps to debris-induced damage in a Condition Report on October 22, 2002. Since that time, FENOC has performed numerous evaluations to determine the impact of the containment emergency sump debris and to identify possible corrective actions. These have included contacting the pump vendor to determine whether the debris would potentially affect pump performance, evaluation of debris transport to the containment emergency sump to determine whether it would contain sufficient debris to potentially affect the pumps during a small-break LOCA, evaluation of various alternatives for correcting the condition, and evaluation of whether the pumps would be required to take suction from the containment emergency sump (via the LPI pumps) in Mode 3 given the low amount of decay heat currently being generated by the reactor core. The results of the latter evaluation were recently provided to FENOC in the enclosed Framatome ANP evaluation, and have formed the basis for this license amendment application. In summary, FENOC has made best efforts to avoid the need for the license amendment, and has prepared this application in a timely manner once it was determined that the amendment would be needed and it received the necessary supporting analysis from its vendor.

In summary, FENOC believes this amendment application satisfies the criteria for an exigent amendment in Section 50.91(a)(6).

8.0 REFERENCES

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment No. 254.

2. "Davis-Besse Restart Test Plan," Revision 2, January 9, 2003.

3. DBNPS Updated Safety Analysis Report through Revision No. 23.

4. Framatome Report 51-5026803-00, "DB-1 Low Decay Heat Mode 3 LOCA Evaluation," April 2003.

5. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

6. EPRI TR-105396, "PRA Applications Guide," August 1995.

7. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. AEC, 1962.

8. Framatome Report 86-5015253, "Davis Besse Cycle 14 Radiation Analysis Report."

9.0 ATTACHMENTS

1. Proposed Mark-Up of Technical Specification Pages

2. Proposed Retyped Technical Specification Pages

3. Technical Specification Bases Pages

LAR 03-0008
Attachment 1

PROPOSED MARK-UP
OF
TECHNICAL SPECIFICATION PAGES

(4 pages follow)

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 280°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,*

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one HPI train inoperable, restore the inoperable HPI train to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one LPI train or its associated decay heat cooler inoperable, restore the inoperable equipment to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once each REFUELING INTERVAL, or prior to operation after ECCS piping has been drained by verifying that the ECCS piping is full of water by venting the ECCS pump casings and discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment emergency sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. For all areas of containment affected by an entry, at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT INTEGRITY is established.

d. At least once each REFUELING INTERVAL by:

   1. Verifying that the interlocks:

      a) Close DH-11 and DH-12 and deenergize the pressurizer heaters, if either DH-11 or DH-12 is open and a simulated reactor coolant system pressure which is greater than the Allowable Value (<328 psig) is applied. The interlock to close DH-11 and/or DH-12 is not required if the valve is closed and 480 V AC power is disconnected from its motor operators.

      b) Prevent the opening of DH-11 and DH-12 when a simulated or actual reactor coolant system pressure which is greater than the Allowable Value (<328 psig) is applied.

   2. a) A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

      b) Verifying that on a Borated Water Storage Tank (BWST) Low-Low Level interlock trip, with the motor operators for the BWST outlet isolation valves and the containment emergency sump recirculation valves energized, the BWST Outlet Valve HV-DH7A (HV-DH7B) automatically close in ≤ 75 seconds after the operator manually pushes the control switch to open the Containment Emergency Sump Valve HV-DH9A (HV-DH9B) which should be verified to open in ≤ 75 seconds.

   3. Deleted

4. Verifying that a minimum of 290 cubic feet of trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

5. Deleted

6. Deleted

e. At least once each REFUELING INTERVAL, by

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

   2. Verifying that each HPI and LPI pump starts automatically upon receipt of a SFAS test signal.

f. By performing a vacuum leakage rate test of the watertight enclosure for valves DH-11 and DH-12 that assures the motor operators on valves DH-11 and DH-12 will not be flooded for at least 7 days following a LOCA:

   1. At least once per 18 months.

   2. After each opening of the watertight enclosure.

   3. After any maintenance on or modification to the watertight enclosure which could affect its integrity.

   The inspection port on the watertight enclosure may be opened without requiring performance of the vacuum leakage rate test, to perform inspections. After use, the inspection port must be verified as closed in its correct position. Provisions of TS 3.0.3 are not applicable during these inspections.

g. By verifying the correct position of each mechanical position stop for valves DH-14A and DH-14B.

   1. Within 4 hours following completion of the opening of the valves to their mechanical position stop or following completion of maintenance on the valve when the LPI system is required to be OPERABLE.

   2. At least once each REFUELING INTERVAL.

h. By performing a flow balance test, during shutdown, following completion of modifications to the HPI or LPI subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

HPI System - Single Pump

Injection Leg 1-1	≥ 375 gpm at 400 psig*
Injection Leg 1-2	≥ 375 gpm at 400 psig*
Injection Leg 2-1	≥ 375 gpm at 400 psig*
Injection Leg 2-2	≥ 375 gpm at 400 psig*

LPI System - Single Pump

Injection Leg 1	≥ 2650 gpm at 100 psig**
Injection Leg 2	≥ 2650 gpm at 100 psig**

* Reactor coolant pressure at the HPI nozzle in the reactor coolant pump discharge.

** Reactor coolant pressure at the core flood nozzle on the reactor vessel.

LAR 03-0008
Attachment 2

PROPOSED RETYPED
TECHNICAL SPECIFICATION PAGE

(1 page follows)

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 280°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump, *

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3 *.

ACTION:

a. With one HPI train inoperable, restore the inoperable HPI train to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one LPI train or its associated decay heat cooler inoperable, restore the inoperable equipment to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

---

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

TECHNICAL SPECIFICATION BASES PAGES

(4 pages follow)

Note: The Bases pages are provided for information only.

3/4.5.1 CORE FLOODING TANKS

The OPERABILITY of each core flooding tank ensures that a sufficient volume of borated water will be immediately forced into the reactor vessel in the event the RCS pressure falls below the pressure of the tanks. This initial surge of water into the vessel provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on volume, boron concentration and pressure ensure that the assumptions used for core flooding tank injection in the safety analysis are met.

The tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The one hour limit for operation with a core flooding tank (CFT) inoperable for reasons other than boron concentration not within limits minimizes the time the plant is exposed to a possible LOCA event occurring with failure of a CFT, which may result in unacceptable peak cladding temperatures.

With boron concentration for one CFT not within limits, the condition must be corrected within 72 hours. The 72 hour limit was developed considering that the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFTs is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of both CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection.

The completion times to bring the plant to a MODE in which the Limiting Condition for Operation (LCO) does not apply are reasonable based on operating experience. The completion times allow plant conditions to be changed in an orderly manner and without challenging plant systems.

CFT boron concentration sampling within 6 hours after an 80 gallon volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the borated water storage tank (BWST), because the water contained in the BWST is within CFT boron concentration requirements.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The operability of two independent ECCS subsystems with RCS average temperature ≥ 280°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Each ECCS subsystem consists of one High Pressure Injection (HPI) train, one Low Pressure Injection (LPI) train (including the associated decay heat cooler), and the necessary piping, valves, instrumentation and controls to provide the required flowpaths from the Borated Water Storage

Tank (BWST) or the Containment Emergency Sump to the reactor vessel. Either subsystem operating in conjunction with the core flooding tanks is capable of supplying sufficient core cooling to maintain the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. With RCS average temperature ≥ 280°F, the Limiting Condition for Operation (LCO) requires the OPERABILITY of a number of independent trains, the inoperability of one component in a train does not necessarily render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this LCO is to maintain a combination of equipment such that 100% of the safety injection flow equivalent to 100% of a single subsystem remains available. This allows increased flexibility in plant operations under circumstances when components in opposite subsystems are inoperable.

With one or more components inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS subsystem is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

The function of the trisodium phosphate dodecahydrate (TSP) contained in baskets located in the containment normal sump or on the 565' elevation of containment adjacent to the normal sump, is to neutralize the acidity of the post-LOCA borated water mixture during containment emergency sump recirculation. The borated water storage tank (BWST) borated water has a nominal pH value of approximately 5. Raising the borated water mixture to a pH value of 7 will ensure that chloride stress corrosion does not occur in austenitic stainless steels in the event that chloride levels increase as a result of contamination on the surfaces of the reactor containment building. Also, a pH of 7 is assumed for the containment emergency sump for iodine retention and removal post-LOCA by the containment spray system.

The Surveillance Requirement (SR) associated with TSP ensures that the minimum required volume of TSP is stored in the baskets. The minimum required volume of TSP is the volume that will achieve a post-LOCA borated water mixture pH of $\geq$ 7.0, conservatively considering the maximum possible sump water volume and the maximum possible boron concentration. The amount of TSP required is based on the mass of TSP needed to achieve the required pH. However, a required volume is verified by the SR, rather than the mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP (53 lb/ft^3). Since TSP can have a tendency to agglomerate from high humidity in the containment, the density may increase and the volume decrease during normal plant operation, however, solubility characteristics are not expected to change. Therefore, considering possible agglomeration and increase in density, verifying the minimum volume of TSP in containment is conservative with respect to ensuring the capability to achieve the minimum required pH. The minimum required volume of TSP to meet all analytical requirements is 250 ft^3. The surveillance requirement of 290 ft^3 includes 40 ft^3 of spare TSP as margin. Total basket capacity is 325 ft^3.

Decay Heat Removal System valves DH-11 and DH-12 are located in an area that would be flooded following a LOCA. These valves are located in a watertight enclosure to ensure their operability up to seven days following a LOCA. Surveillance Requirements are provided to verify the acceptable leak tightness of this enclosure. An inspection port is located on this watertight enclosure, which is typically used for performing inspections inside the enclosure. During the vacuum leakage rate test, the inspection port is in a closed position and subject to the test. This inspection port may be subsequently opened for use in viewing inside the enclosure. Opening this inspection port will not require performance of the vacuum leakage rate test because of the design of the closure fitting, which will preclude leakage under LOCA conditions, when properly installed. Proper installation includes independent verification.

Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Containment Emergency Sump Recirculation Valves DH-9A and DH-9B are de-energized during MODES 1, 2, 3 and 4 to preclude postulated inadvertent opening of the valves in the event of a Control Room fire, which could result in draining the Borated Water Storage Tank to the Containment Emergency Sump and the loss of this water source for normal plant shutdown. Re-energization of DH-9A and DH-9B is permitted on an intermittent basis during MODES 1, 2, 3 and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent closure of the valves in the event of a fire, which could result in a loss of the availability of the BWST. Re-energization of valves DH-7A and DH-7B is permitted on an intermittent basis during MODES 1, 2, 3, and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

The Decay Heat Isolation Valve and Pressurizer Heater Interlock setpoint is based on preventing over-pressurization of the Decay Heat Removal System normal suction line piping. The value stated is the RCS pressure at the sensing instrument's tap. It has been adjusted to reflect the elevation difference between the sensor's location and the pipe of concern.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on the BWST minimum volume (500,100 gallons of borated water, conservatively rounded up from the calculated value of 500,051 gallons) and boron concentration ensure that:

1) sufficient water is available within containment to permit recirculation cooling flow to the core following manual switchover to the recirculation mode, and

Docket Number 50-346
License Number NPF-3
Serial Number 2950
Enclosure 2

FRAMATOME REPORT NO. 51-5026803-00

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ENGINEERING INFORMATION RECORD

Document Identifier 51 - 5026803- 00

Title DB-1 LOW DECAY HEAT MODE 3 LOCA EVALUATION

PREPARED BY:		REVIEWED BY:	
Name J. A. KLINGENFUS		Name G. J. WISSINGER	
Signature	Date 5/7/03	Signature	Date 5/7/03

Technical Manager Statement: Initials

Reviewer is Independent.

Remarks:

The original HPI pumps installed in the Davis-Besse Nuclear Power Station Unit No. 1 (DB-1) have recently been evaluated for debris tolerance during the long-term sump recirculation phase. The evaluation raised concerns that debris could lodge and plug the small clearances in the cooling flow paths of the water lubricated hydrostatic bearings, potentially compromising the continued operation of the high pressure injection (HPI) pumps during sump recirculation phase. When the HPI suction source is from the borated water storage tank (BWST), there is no source of debris and the pump operation is not in question. However, debris could collect and impair operation of the HPI pump after the suction source is switched from the BWST to the discharge of a low pressure injection (LPI) pump that is taking suction from the containment emergency sump. As a result of these vulnerabilities, the FirstEnergy Nuclear Operating Company (FENOC) is taking corrective action to ensure the HPI pumps at DB-1 will be operable during the sump recirculation phase before the plant returns to power operation. In the meantime, FENOC would like to proceed with the special seven-day Mode 3 NOP/NOT leak test with the current HPI pumps. FANP has been tasked with evaluating the post-LOCA core cooling capabilities without credit for HPI flow during the recirculation phase to support this test.

A representative SBLOCA analysis and related LOCA evaluations were completed to demonstrate adequate core cooling performance for the entire spectrum of required break sizes and locations. It was concluded that this low decay heat Mode 3 NOP/NOT test could be performed with the current HPI pumps installed at DB-1 without HPI flow credit during the sump recirculation phase. Some unique procedural guidance with alternate core cooling flow paths have been developed for this test and are provided in Section 5 to successfully mitigate all LOCA break sizes and locations with the limiting single failure, without credit for HPI flow during the sump recirculation phase. These additional operator actions are recommended for consideration several hours into the event as directed by the technical support center (TSC) staff in the unlikely event that a LOCA occurs during this NOP/NOT test. Therefore, it is concluded that it is acceptable for FENOC to proceed with the Mode 3 NOP/NOT test with the current HPI pumps.

Record of Revisions

Revision Number/Date	Change Section/Paragraph	Description
00 / May 2003	All	Initial release

DB-1 LOW DECAY HEAT MODE 3 LOCA EVALUATION

Executive Summary

The original HPI pumps installed in the Davis-Besse Nuclear Power Station Unit No. 1 (DB-1) have recently been evaluated for debris tolerance during the long-term sump recirculation phase. The evaluation raised concerns that debris could lodge and plug the small clearances in the cooling flow paths of the water lubricated hydrostatic bearings, potentially compromising the continued operation of the high pressure injection (HPI) pumps during sump recirculation phase. When the HPI suction source is from the borated water storage tank (BWST), there is no source of debris and the pump operation is not in question. However, debris could collect and impair operation of the HPI pump after the suction source is switched from the BWST to the discharge of a low pressure injection (LPI) pump that is taking suction from the containment emergency sump. As a result of these vulnerabilities, the FirstEnergy Nuclear Operating Company (FENOC) is taking corrective action to ensure the HPI pumps at DB-1 will be operable during the sump recirculation phase before the plant returns to power operation. In the meantime, FENOC would like to proceed with the special seven-day Mode 3 NOP/NOT leak test with the current HPI pumps. FANP has been tasked with evaluating the post-LOCA core cooling capabilities without credit for HPI flow during the recirculation phase to support this test.

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Table of Contents

List of Tables

List of Figures

List of Commonly Used Acronyms

AFPT	– Auxiliary Feedwater Pump Turbine
AFW	– Auxiliary Feedwater
Appendix K	– 10 CFR 50 Appendix K
APS	– Auxiliary Pressurizer Spray
AVV	– Secondary Side Atmospheric Vent Valve
BCM	– Boiler Condenser Mode
BWST	– Borated Water Storage Tank
CFR	– Code of Federal Regulations
CFT	– Core Flood Tank
CLPD	– Cold Leg Pump Discharge
CLPS	– Cold Leg Pump Suction
CTMT	– Containment
DB-1	– Davis-Besse Nuclear Power Station Unit 1
DHR	– Decay Heat Removal
ECCS	– Emergency Core Cooling System
EM	– Evaluation Model
EOPs	– Emergency Operating Procedures
FANP	– Framatome ANP Incorporated
FENOC	– First Energy Nuclear Operating Company
FP	– Full Power
GDC	– 10 CFR 50 Appendix A General Design Criteria
HPI	– High Pressure Injection
HPV	– High Point Vents
HZP	– Hot, Zero Power
IMI	– Incore Measurement Instrumentation
LBLOCA	– Large Break Loss of Coolant Accident
LOCA	– Loss of Coolant Accident
LOFW	– Loss of Feedwater
LSCM	– Loss of Subcooling Margin
LPI	– Low Pressure Injection
MDFP	– Motor-Driven Feedwater Pump
MFW	– Main Feedwater
MSIV	– Main Steam Isolation Valve
MSSV	– Main Steam Safety Valve
MU	– Makeup
MUP	– Makeup Pump
NC	– Natural Circulation
NOP	– Normal Operating Pressure
NOT	– Normal Operating Temperature
OTSG	– Once Through Steam Generator
PCT	– Peak Cladding Temperature
PORV	– Pilot-Operated Relief Valve

List of Commonly Used Acronyms (Continued)

PSV	– Pressurizer Code Safety Valve
PTS	– Pressurized Thermal Shock
PZR	– Pressurizer
RC	– Reactor Coolant
RCP	– Reactor Coolant Pump
RCS	– Reactor Coolant System
RV	– Reactor Vessel
SBLOCA	– Small Break Loss of Coolant Accident
SCM	– Subcooling Margin
SFAS	– Safety Features Actuation System
SFRCS	– Steam & Feedwater Rupture Control System
SG	– Steam Generator

DB-1 LOW DECAY HEAT MODE 3 LOCA EVALUATION

1. Introduction

The Davis-Besse Nuclear Power Station Unit No. 1 (DB-1) shut down on February 16, 2002. As part of the plant restart activities, the plant will heat up and maintain normal operating pressure (NOP) and near normal operating temperatures (NOT) for a special seven-day system leak test. The RCPs will be used to heat the system up to Mode 3 NOP/NOT conditions. After seven days the plant will cool down and perform additional inspections for any indication of leakage.

The original HPI pumps installed in the DB-1 plant have recently been evaluated for debris tolerance during the long-term sump recirculation phase. The evaluation raised concerns that debris could lodge and plug the small clearances in the cooling flow paths of the water lubricated hydrostatic bearings, potentially compromising the continued operation of the HPI pumps during the sump recirculation phase. When the HPI suction source is from the BWST, there is no source of debris and the pump operation is not in question. However, debris could collect and impair operation of the HPI pump after the suction source is switched from the BWST to the discharge of an LPI pump that is taking suction from the containment emergency sump. As a result of these vulnerabilities, FENOC is taking corrective action to ensure the HPI pumps at DB-1 will be operable during the sump recirculation phase before the plant returns to power operation. In the meantime, FENOC would like to proceed with the Mode 3 NOP/NOT leak test with the current HPI pumps. FANP has been tasked with evaluating the post-LOCA core cooling capabilities without credit for HPI flow during the recirculation phase.

In the unlikely event that a LOCA should occur during this leak test, the ECCS systems will activate to mitigate the consequences of the event. The HPI, CFT, and LPI systems work collectively (possibly with the non-safety makeup pumps when taking suction from the BWST) to supply ECCS to the RCS. The secondary side feedwater systems (both main and auxiliary feedwater from the turbine driven AFW pumps) will provide the necessary flow to remove RCS energy not removed by the break. Because the core power is extremely low, the ECCS and AFW systems would have little difficulty keeping the core cool provided the pump operation credited in the full power LOCA analyses is preserved. However, in light of recent issues, the operability of the current HPI pumps during the sump recirculation phase has been brought into question. Mode 3 operation without the HPI pumps during the recirculation phase is not consistent with the description in the DB-1 Updated Safety Analysis Report. FENOC is planning to submit a license amendment application based on FANP analyses and evaluations that show the Mode 3 test can be performed safely with full compliance to the 10 CFR 50.46 acceptance criteria without credit for the HPI pumps during the sump recirculation phase.

The task of evaluating the 10 CFR 50.46 acceptance criteria under the low decay heat power conditions will consider the LOCA design basis HPI functions that may be needed during the sump recirculation phase for any postulated LOCA. The HPI pumps have been relied upon during the post-LOCA long-term sump recirculation phase to perform the following functions.

1. At least one HPI pump must supply ECCS flow when the RCS pressure is above the pressure at which LPI flow is delivered to the RCS.

2. At least one HPI pump must supply ECCS flow if only one LPI pump is providing ECCS flow and is not providing at least 1000 gpm of flow to each CFT nozzle per the generic EOPs (Reference 8). (Note: The 1000 gpm flow is based on matching the core decay from full power operation plus additional ECCS to ensure abundant core cooling per 10 CFR 50 Appendix A GDC 35.)

3. The HPI pump must supply flow to the auxiliary pressurizer spray line for use in core boron precipitation control if needed.

The evaluations herein identify normal plant cooldown methods with non-safety equipment that can and will be used in the LOCA mitigation. However, the LOCA evaluations performed to demonstrate compliance with the 10 CFR 50.46 acceptance criteria (without credit for the HPI pumps during the sump recirculation phase) credit only safety-grade equipment availability and consider any limiting single failure. These evaluations were performed without extensive use of the RELAP5/MOD2 Evaluation Model (BWNT LOCA EM, Reference 5) to show that the Mode 3 LOCA PCT consequences are in compliance with 50.46 and they are bounded by the current full power LOCA analyses of record. Demonstrating compliance with the 10 CFR 50.46 acceptance criteria is not in question during the BWST draining period, however, special low power Mode 3 operator guidance is provided and credited in the evaluation of the LOCA mitigation during the long-term without HPI pumps during the sump recirculation phase. This LOCA mitigation guidance includes provisions for use of all equipment that may be available (safety or non-safety), although 50.46 compliance is based on only safety-related equipment.

2. ECCS Requirements and System Description

2.1. ECCS Requirements

A loss-of-coolant accident (LOCA) is a hypothetical accident that results in the loss of reactor coolant from breaks in pipes in the reactor coolant pressure boundary at a rate in excess of the capability of the reactor coolant makeup system up to and including a double-ended rupture of the largest pipe in the reactor coolant system. The Code of Federal Regulations (CFR) requires each pressurized light-water nuclear power reactor to have an emergency core cooling system (ECCS) that is designed so that its calculated cooling performance following a postulated LOCA conforms to the criteria set forth in 10 CFR 50.46. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. The evaluation methods must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA.

The five ECCS criteria contained in 10 CFR 50.46 for a commercial nuclear power plant are:

1. The calculated peak cladding temperature (PCT) is less than 2200 F.

2. The maximum calculated local cladding oxidation is less than 17.0 percent.

3. The maximum amount of core-wide oxidation does not exceed 1.0 percent of the fuel cladding.

4. The core remains amenable to cooling.

5. Long-term cooling is established and maintained after the LOCA.

Framatome ANP has demonstrated that the DB-1 plant meets the 10 CFR 50.46 requirements by analyzing the limiting pipe break loss-of-coolant accidents (LOCAs) (Reference 7) with an NRC-approved evaluation model (EM) (Reference 5). The limiting breaks are generally those that result in the largest bypass of emergency core cooling system (ECCS) flow directly out of the break. The break size range includes any break that can exceed the makeup system flow up to and including that of a full, double-ended guillotine rupture of the cold leg or hot leg pipe. The mitigation of the break consequences is accomplished by a combination of safety injection flow, RCS heat removal via auxiliary (or emergency) feedwater (AFW), and long-term cooling via decay heat coolers with ECCS recirculation from the containment sump. These systems are activated and managed by both automatic trips and controls, plus manual operator actions identified in the plant emergency operating procedures (EOPs).

2.2. ECCS and AFW System Descriptions

Successful LOCA mitigation for the entire spectrum of break sizes uses combinations of the different ECCS systems and AFW heat removal. The ECCS flows supply safety injection to make up for RCS mass loss, core and stored metal heat sources, and flashing. The AFW injection to the secondary side provides RCS heat removal when the break size is insufficient to pass all of the core or RCP generated energy addition.

2.2.1. ECCS System

The ECCS equipment used specifically for LOCA mitigation at the DB-1-1 plant consists of the combination of two intermediate head high pressure injection (HPI) pumps, two nitrogen-pressurized core flood tanks (CFTs), and two low pressure injection (LPI) pumps. The piping leading from the HPI pumps connects to the cold leg pump discharge (CLPD) piping just upstream of the reactor vessel. Each HPI pump can provide flow to two of the four CLPD pipes. The CFT and LPI piping tee into a common line that enters directly into the reactor vessel (RV) downcomer slightly above the inlet and exit nozzle belt region. In addition, the DB-1 plant has two makeup pumps that are used in the makeup and purification system during normal operation. They are capable of injecting flow into the RCS at pressures up to and including the maximum setpoint of the pressurizer code safety relief valves. The makeup pumps can augment the ECCS flow significantly, however, their classification is such that they are not typically used in providing credited ECCS flow to the RCS for LOCA mitigation.

The HPI system is designed to provide flow up to moderately high RCS pressures and is most critical for small break LOCA (SBLOCA) mitigation. It also provides redundant ECCS injection locations for larger LOCAs and post-LOCA boron concentration control through the auxiliary pressurizer spray line. The HPI system is arranged in two separate trains with independent power sources. The pumps are activated automatically after receiving a safety features actuation signal (SFAS). For LOCAs, automatic SFAS occurs on low RCS pressure with a backup trip based on high containment pressure. Alternatively, the operators may initiate SFAS on indication of a loss of adequate subcooling margin (LSCM), although this action is typically not credited in LOCA analyses. Once activated, the HPI pumps provide flow from the borated water storage tank (BWST) when the RCS pressure is below the shutoff head of the pumps. A recirculation line to the BWST provides an acceptable minimum pump flow during the BWST flow period when the RCS pressure is above the shutoff head of the HPI pump. There is sufficient redundancy of equipment and flow alignments such that at least one full train remains operative under the assumption of a single active failure.

There are two 1400 ft^3 core flood tanks, each containing approximately 1000 ft^3 of borated water and 400 ft^3 of nitrogen gas nominally pressurized to 600 psig. Each tank connects individually to the RV upper downcomer region approximately 1.5 ft above the nozzle belt centerline. In-line check valves prevent reverse flow during normal operation. The system is, therefore, self-contained, self-actuating, and passive. Flow into the RCS occurs whenever the RCS pressure falls below the tank pressure. The CFTs are most critical for refilling the RV lower plenum and lower core for intermediate to large LOCA break sizes. After the tanks empty, the operators are instructed to isolate the tanks if possible to prevent injection of the nitrogen into the RCS.

Low-pressure injection is achieved with the decay heat removal (DHR) system pumps. There are two pumps in independent trains each with separate power supplies. Normally used for cooling when the reactor is not operating, the system also serves the low-pressure ECC injection function by providing borated water through the two CFT injection nozzles. In emergency operation, the DHR (or commonly referred to as LPI) pumps initially inject water from the BWST. When the BWST low-level set point is reached, the operators align the LPI pumps to take suction from the containment emergency sump. During sump recirculation, injection flow is passed through a heat exchanger before returning to the RCS. The system contains sufficient redundancy such that one full train is available under a single active failure. Automatic actuation is provided by SFAS on low-low RCS pressure or high containment pressure. The LPI pumps are critical for continuing the core refill following LBLOCA, providing sump recirculation for all LOCAs, providing post-LOCA boron concentration control through alternate injection or suction alignments, and providing long-term core cooling in the sump recirculation or DHR modes.

In the recirculation mode, the LPI pumps are the only ECCS pumps capable of taking suction from the containment sump. Long-term, high pressure cooling is possible because the HPI pumps can take suction from the LPI pump discharge and deliver coolant through their cold leg connections to the RCS. This allows the HPI pumps to provide ECCS injection when the RCS pressure remains above the LPI pressure range, i.e., above 215 psia. It also provides redundant ECCS injection locations. (For DB-1, FENOC Condition Report 02-08492 identifies that the continuous capability to provide for HPI suction during the sump recirculation phase from the containment (CTMT) emergency sump could be compromised due to the CTMT sump debris transport to the HPI pumps.)

The variety of ECCS piping arrangements allows the safety injection flow to be distributed to multiple injection locations. The redundancy of injection locations is critical to ensuring that a substantial fraction of the ECCS will reach the core and makeup for the boiloff from the core heat generation. A minimum of two injection locations with individual line flow indication, or three flow paths without flow indication may be required. When flow indication is available, two paths are sufficient because the break location could bypass only the ECCS flow provided to a single injection location. If flow indication is not available, at least three injection locations may be needed to ensure ECCS reaches the core. With the three potential injection paths, the break could bypass the flow from one injection location, a single failure could possibly prevent one flow path from opening, while the flow from the third line reaches the core.

2.2.2. AFW System

LOCA analyses typically assume the accident occurs when the plant has been operating at full power. Under these conditions, the event results in the loss of RCS liquid inventory at a rate determined by the break area and fluid conditions just upstream of the break location. The net energy and mass changes to the system determine the RCS pressure response during a LOCA. The core heat generation and RCP energy addition rates are the dominant initial heat sources for the RCS. After the reactor trips, the RCPs trip, and the RCS depressurization begins, the fuel stored energy and passive metal stored energy release rates are other sources of energy. The break energy relief and steam generator heat removal are the net energy sinks for the RCS. The RCS pressure response to the LOCA determines how rapidly SFAS occurs and the time and rate at which the ECCS systems inject liquid to makeup for the break mass discharge and other liquid losses through core and passive metal boiloff rates and liquid flashing contributions. The break size and location along with the ECCS injection rates control the RCS inventory distribution and determine how much steam generator heat removal is necessary. The range of LOCA break sizes not only tests the capability of the ECCS systems to deliver flow to the RCS, but also places reliance on the auxiliary feedwater (AFW) systems for smaller break sizes. Because the DB-1 plant has intermediate head HPI pumps, the AFW injection is critical to providing the steam generator heat removal that will reduce the RCS pressure below the shutoff head of the HPI pumps.

The DB-1 plant has two turbine-driven AFW pumps and a backup motor-driven AFW pump with liquid supplied from the Condensate Storage Tanks, or, if necessary, from the Service Water system. The AFW pumps supply liquid to nozzles near the top of the SG secondary side tube bundle where it wets the tubes on the periphery of the tube bundle and removes energy from the RCS. The RCS energy removal controls the RCS pressure to ranges where the ECCS injection is effective. Although the AFW system is not strictly part of the ECCS, it is a required safety system needed to mitigate small LOCAs. The DB-1 plant typically credits the turbine-driven AFW pump(s) in the LOCA analyses. There is suitable redundancy of equipment and flow paths to assure that AFW is available to at least one steam generator for any postulated single failure of equipment or power supplies.

3. DB-1 Mode 3 LOCA Considerations and Initial Conditions

3.1. DB-1 Mode 3 Test Plans

The DB-1 nuclear power plant has been shut down since February 16, 2002. As part of the plant restart activities, the plant will heat up and maintain normal operating pressure (NOP) and near normal operating temperatures (NOT) for a seven-day system leak test. The RCPs will be used to heat the system up to Mode 3 NOP/NOT conditions. After seven days DB-1 will cooldown and perform additional inspections for any indication of leakage.

In the unlikely event that a LOCA should occur during this leak test, the ECCS systems would activate and perform their safety functions to mitigate the consequences of the event. The HPI, CFT, and LPI systems work collectively (possibly with the non-credited makeup pumps) to supply ECCS to the RCS. The secondary side feedwater systems (both main and auxiliary feedwater from the turbine driven AFW pumps) would provide the necessary flow to remove RCS energy not removed by the break. Because the core power is extremely low, the ECCS and AFW systems would have little difficulty keeping the core cool when utilizing the continuous pump flows (from the BWST and sump recirculation phases) credited in the full power LOCA analyses.

The problem for the upcoming Mode 3 test is the long-term operability of the HPI pumps because of debris intolerance during the sump recirculation phase. When the HPI suction source is from the BWST, there is no source of debris and the pump operation is not in question. However, debris could collect and impair operation of the HPI pump after the suction source is switched from the BWST to the discharge of an LPI pump that is taking suction from the containment emergency sump. Because continued HPI operation during sump recirculation is not assured with the current HPI pumps, FANP has been tasked with evaluating the post-LOCA core cooling capabilities without credit for HPI flow during the recirculation phase for this low core decay heat Mode 3 test.

The task of evaluating the 10 CFR 50.46 acceptance criteria under the low decay heat power conditions was completed by reviewing the HPI functions that may be needed during the sump recirculation phase for any postulated LOCA. It was determined that flow from the HPI pumps may be required during the post-LOCA long-term sump recirculation phase to perform the following functions.

1. The HPI pumps must supply ECCS flow to at least two cold legs if the RCS pressure is above the pressure at which LPI flow can be obtained, 215 psia. The HPI flow from one pump (when only one HPI pump is in operation from single failure criterion) must be split to two cold legs, with each leg having an adequate amount of flow, thereby keeping the core from exceeding the acceptance criteria. Each leg must have adequate flow because there is a possibility that one leg could be flowing to a broken HPI line or cold leg with a break between the

injection location and the core. Any flow going to the broken line would flow out of the break into containment and not be available for core cooling.

2. The HPI pumps must supply ECCS flow if only one LPI pump is providing ECCS flow and either the LPI crosstie line has not been opened or the flow has not been balanced to provide at least 1000 gpm of flow to each CFT nozzle. In these cases the HPI flow must supply adequate ECCS flow because the one LPI pump may be injecting into the broken CFT line.

3. The HPI pump must supply flow to the auxiliary pressurizer spray line for use in core boron precipitation control.

The LOCA evaluations contained herein assume the use of only safety-grade equipment and included any limiting single failure to demonstrate that the 10 CFR 50.46 acceptance criteria can be met without credit for the HPI pumps during the sump recirculation phase. The evaluations used RELAP5/MOD2 Evaluation Model (BWNT LOCA EM, Reference 5) analyses as necessary to validate any break scenarios that cannot be readily shown to be less limiting than a similar LOCA break size from the full power analyses. Operator guidance to mitigate the event is also provided that includes provisions for use of all equipment that may be available (safety or non-safety).

3.2. DB-1 Mode 3 Test Plant Initial Conditions

3.2.1. Plant Initial Conditions

The Mode 3 NOP/NOT leak test will use present decay heat and RCPs to heat the system up to target Mode 3 conditions of 532 F and 2170 psia. Makeup, letdown, pressurizer sprays, and pressurizer heaters will be used to control the RCS inventory and pressure during the heatup. The operators will control MFW to a conservatively low uncertainty adjusted SG secondary side level of 30 inches to remove the RCP heat addition. The MFW temperature will be approximately 186 F. The turbine bypass line will be opened as necessary to control the secondary side pressure to roughly 850 psig. It is important to note that, because the secondary side pressure is below the primary side saturation pressure, it will not become a heat source unless the RCS is depressurized. For this Mode 3 test, the control rods will either be fully inserted with trip breakers open, or a single safety group of control rods will be withdrawn to provide a prompt source of trippable negative reactivity. Nonetheless, the RCS boric acid concentration will be maintained at or above the all rods out (ARO) critical boron concentration. Therefore, the core power will remain essentially constant at the current decay heat level.

For the Mode 3 test, it is highly probable that offsite power will not be interrupted. Since the DB-1 plant is not producing power the in-house plants loads are already being supplied from the Startup Transformer and the need for the automatic transfer of electrical loads is negated. This means that there would be no automatic transfer from the Auxiliary Transformer to the Startup Transformer(s) following a LOCA and subsequent turbine trip. There would also be no grid perturbation from a LOCA-induced turbine trip that could potentially cause the grid to become unstable and result in a loss of offsite power. Nonetheless, even though offsite power should be available, the 10 CFR 50.46 acceptance criteria must be met with or without credit for offsite power.

Table 3-1 provides a summary of the test conditions considered in the LOCA evaluations and analysis. It also identifies critical operator actions required for successful mitigation of a LOCA occurring at these test conditions.

Table 3-1. Low Power Mode 3 LOCA Inputs and Assumptions for DB-1

Parameter	Value	Key Inputs
RCS Initial Conditions		
Irradiated Fuel Normalized Power Level (P/P_o) 1.2 * ANS71 w/B&W Heavy Actinides	0.00089 (403 days after shutdown)	Yes
Number of Fresh Assemblies in Core	76	Yes
Number of Irradiated Assemblies in Core	101	Yes
Core Power Level Evaluated	1.44 MWt (2772*1.02*0.00089*101/177)	Yes
Decay Heat (Constant during LOCA Transient)	1.44 MWt	Yes
Initial PZR Level, inches above lower tap	220	
Hot Leg Tap Pressure, psia	2170	Yes
Average RCS Temperature, F	≤ 532	Yes
RCP Heat, MWt	20 (net 16 with MU and LD)	
Letdown Flow, gpm	70	
Initial RCS Flow, gpm	390,000	
Initial Steam Generator Pressure, psia	864	
MFW Temperature, F	186	
CFT Initial Pressure, psia	615 ± 33 (Minimum used)	Yes
CFT Initial Volume, ft^3	1040 ± 40 (Maximum used)	Yes
Transient Parameters and Boundary Conditions		
Pressurizer Heaters and Sprays for Level Control	Not used in LOCA analyses	
Low RC Pressure Rx Trip Setpoint, psia	1900 (Used for TSV closure)	
Secondary Main Steam Flow Isolation	Reactor Trip + 0.5 seconds	
Low RC Pressure Trip Setpoint, psia	1515	Yes
Low RC Pressure HPI Delay, sec	60.0	
Loss of RCS Subcooling Margin RCP Trip, F	20.0	Yes
AFW Initiation Delay on RCP Trip, sec	60.0	
SG LSCM Level Setpoint, in	106 (8.833 ft)	
BWST Temperature, F	90	
LPI Auxiliary Pressurizer Spray Flow, gpm	≥40 gpm	Yes
AFW Temperature, F	120	
AFW Flow Rate, gpm (1 pump)	See Table 3-4 (Ref. 7)	
HPI Flow Rate, gpm (1 pump) From BWST only	See Tables 3-6, 3-7, 3-8 (Ref. 7)	Yes
LPI Flow Rate, gpm (1 pump) From BWST and Sump	See Table 3-5 (Ref. 7)	Yes
Critical Operator Actions		
1. Trip RCPs on LSCM		Yes
2. Ensure initiation and appropriate SG level control with AFW		Yes
3. Ensure initiation of at least 1 HPI pump and control flow splits as identified in Reference 7		Yes
4. Ensure initiation of at least 1 LPI pump with flows given in Reference 7		Yes
5. Depressurize RCS if necessary to LPI pressure before BWST switchover		Yes [1]
6. Complete BWST to sump switchover for the LPI pump suction source		Yes
7. Open LPI Crosstie line to get flow to both CFT lines or initiate LPI APS		Yes [2]
8. Initiate post-LOCA core boron concentration control as necessary with LPI APS		Yes [3]
9. Perform necessary actions or maintenance to preserve LPI flow during sump recirculation		Yes

Notes for Table 3-1 provided on next page.

Notes for Table 3-1:

1. Operator Action 5: This action is for very small break sizes after the RCS pressure finds an equilibrium well above pressures that allow any LPI injection. RCS depressurization can be accomplished in several ways with non-safety grade equipment (PORV, SG heat removal) and ECCS throttling to control core exit subcooling when the RCP are in operation, or ECCS throttling when the RCS is not adequately subcooled. Other than throttling HPI without subcooling margin, these actions are procedurally in place at DB-1 and the operators should have performed them. They are not included in LOCA analyses, however, because they involve non-safety related components. The LOCA analyses are typically terminated when the plant is in stable hot conditions, with the core shutdown, and no obvious challenges to maintaining adequate core cooling in the long term. For this Mode 3 test, however, the operators must depressurize the RCS to the LPI flow range to manage the potential that HPI pumps are not available during the sump recirculation phase.

2. Operator Action 7: This operator action is primarily targeted at the CFT line break, or any other larger LOCA that has only LPI flow. At least two confirmed injection locations are needed to assure 100% of the ECCS is not flowing out a broken line to containment. Therefore, the LPI crosstie needs to be opened and flow balanced between the two lines. This action is proceduralized at DB-1 to address abundant cooling but has not been credited in the existing LOCA analyses. If the flow path cannot be established, or flow indication is not available, then LPI flow through the auxiliary pressurizer spray line is a redundant flow path. Another option is flow through the HPI piggy-back line through an inoperable HPI pump with the HPI injection valves open. These alternate flow paths are adequate only for the Mode 3 test low decay heat conditions.

3. Operator Action 8: This operator action is either a modification of the current primary (HPI auxiliary pressurizer spray [APS] method) and secondary (DHDL flow if two LPI pumps are available) boron precipitation methods or another backup that will work because the decay heat is so low.

3.2.2. Core Decay Heat Determination

If HPI is lost during the sump recirculation phase of the event, it is possible that the RCS pressure could be above the CFT and LPI injection pressures, which means that no ECCS would be available to replace liquid lost through the break. Under worst-case assumptions, as the system depressurizes from the break flow, the core could uncover, due to liquid inventory loss, core boiling, and flashing. The core decay heat rate is important to determining the methods and time sequence for the operators to depressurize the RCS to the LPI pressure. It also defines the core heatup rate should core uncovering occur.

The DB-1 plant has been shut down since February 16, 2002. Since that time 76 fresh (Reference 2), unirradiated fuel assemblies have been loaded with 101 previously irradiated assemblies. The core decay heat is quite low. The control rods will either be fully inserted with trip breakers open, or a single safety group of control rods will be withdrawn to provide a prompt source of trippable negative reactivity. Nonetheless, the RCS boric acid concentration will be maintained at or above the all rods out (ARO) critical boron concentration. Therefore, the core power will remain essentially constant at the current decay heat level. The decay heat can be calculated based on a realistic time at operation or the traditional infinite operation required by 10 CFR 50.46. These two calculations provide some information regarding the degree of conservatism in the decay heat for long-times post shutdown.

The realistic decay heat on April 1, 2003 (after 408 days shutdown) is estimated based on the decay heat curve given on page 159 of Reference 3 assuming that the plant had operated continuously for 800 days before shutdown. The normalized power (P/P_0) based on 120 percent fission product decay multiplier for a whole core of irradiated assemblies was 0.00021. Considering that 76 fuel assembles are fresh, the estimate of total core power for 101 of 177 fuel assemblies based on 102 percent of 2772 MWt is calculated to be 0.34 MWt (2772 MWt * 1.02 * 0.00021 * 101 Assemblies / 177 Assemblies).

For the LOCA evaluation, the decay heat used must be calculated based on the ANS 1971 infinite operation fission product decay with a 1.2 multiplier as required by 10 CFR 50 Appendix K. The B&W heavy isotopes actinide model used in the current DB-1 BWNT LOCA EM analyses was also included. At 408 days after shutdown, the time post trip 3.52×10^7 seconds (408 days * 24 hours/day * 3600 seconds/hour). The nearest time value in the Reference 4 decay heat table is 3.4×10^7 seconds, which coincides with 394 days post trip. The normalized decay heat is 0.00089 at this time. A full core of irradiated fuel run at 102 percent of 2772 MWt would have a core power of 2.52 MWt (1.02 * 2772 MWt * 0.00089). This is equivalent to 2390 Btu/sec (2.52 MWt * 948 Btu/s/MWt). Considering that 76 fuel assembles have been replaced with fresh fuel without any decay heat, the total core power is calculated as 1.44 MWt (2.52*101/177) or 1360 Btu/s (2390 *101/177).

At the Appendix K core power, the core average linear heat rate (LHR) for the burned assemblies is calculated as

$$\begin{aligned} LHR_{ave} &= Q_{DH\,tot} / (N_{asmbl} * L_{asmbl} * N_{pins/asmbl}) \\ &= 1440\ kW / (101 * 12\ ft * 208) \\ &= 0.00571\ kW/ft \end{aligned}$$

If the hot spot in the core uncovers, it will heat up at a rate consistent with the maximum core peaking factor, F_Q, times the average linear heat rate limit (LHR). The SBLOCA EM analyses were performed at full power allowed operation at a maximum LHR limit of 17.5 kW/ft (Reference 7). This LHR limit is equivalent to a hot spot normalized peak of

$$\begin{aligned} F_Q &= LHR_{Max@102\%FP} / LHR_{ave@102\%FP} \\ &= LHR_{Max@102\%FP} / [Q_{core@102\%FP} / (N_{asmbl} * L_{asmbl} * N_{pins/asmbl})] \\ &= 17.5\ kW/ft / [(1.02 * 2772000\ kW) / (177 * 12\ ft * 208) \\ &= 2.73 \end{aligned}$$

The hot spot did not run at this power level continuously. Nonetheless, if it had, the hot spot LHR limit after the nearly 14 months of shutdown would be 0.0156 kW/ft (2.73 * 0.00571 kW/ft). Because the peaking represented here is at the limits of normal operation, it is assured that any fuel location has a linear heat rate limit less than this value. The plant cannot stay at the limits of normal operation for more than several hours without having to decrease power; therefore, although it is highly conservative, the allowed F_Q value it is not very realistic for use in the LOCA evaluation. The Cycle 13 Final Fuel Cycle Design (Reference 10) gives typical F_Q for normal steady state distributions in the range of 1.6 to 1.9. The peak pins had normal radial power distributions during the cycle of 1.5. The typical axial power shapes are calculated to be between 1.07 (1.6/1.5) and 1.27 (1.9/1.5). Therefore, a peak of 2.0 was selected as a reasonable maximum peak used to determine the maximum adiabatic heat up for the irradiated assemblies should there be core uncovering. The hot spot LHR limit with this peaking is 0.0114 kW/ft (2.0 * 0.00571 kW/ft).

Typical LOCA analyses from full power that could uncover the core can calculate adiabatic heatup rates in the range of 5 F/second for the SBLOCAs up to a maximum range of 20 to 25 F/second for LBLOCAs. As was shown above, the core power is very low for this NOP/NOT test compared to full power operation. If the core is uncovered at this NOP/NOT linear heat rate limit, it would take a substantial time period for the core to heat up to temperatures that would yield any additional heat from the cladding metal-water reaction. The adiabatic heatup rate is calculated below for use in determining how rapidly the hot spot could heat up during core uncovering at the current Appendix K decay heat rate.

The net heat generation and volumetric heat capacity of a 1-foot section of the pin at the peak LHR limit was used to determine the adiabatic heatup rate. For the purpose of the calculation, the volumetric heat capacity of the gap gasses was excluded and the cold dimensions for the Mark-B10K fuel pin were used. The volumetric heat capacities of the fuel and clad were taken at 1000 F from the DB-1 RELAP5/MOD2 model (Reference 6).

The adiabatic heat up rate, $\Delta T/\Delta t_{adiabtic}$, is calculated using the fuel and cladding volumetric heat capacities and the post trip hot spot LHR limit.

$$\begin{aligned}
\Delta T/\Delta t_{adiabtic} &= [LHR_{DH\text{-}hot\ spot} / (\rho^* c_p * V_{pin/ft})] \\
&= \{LHR_{DH\text{-}hot\ spot} / [(\rho^* c_p * V_{fuel/ft})_{fuel} + (\rho^* c_p * V_{clad/ft})_{clad}]\} \\
&= \{LHR_{DH\text{-}hot\ spot} / [(\rho^* c_p * \pi * r_{fuel}^2)_{fuel} + \\
&\qquad (\rho^* c_p * \pi\, (r_{clad\ od}^2 - r_{clad\ id}^2))_{clad}]\} \\
&= \{0.0114\ kW/ft * 0.948\ Btu/s/kw / \\
&\qquad [(48.10\ Btu/ft^3\text{-}F * \pi * (0.0155625\ ft)^2)_{fuel} \\
&\qquad + (35.52\ Btu/ft^3\text{-}F * \pi * ((0.0179167\ ft)^2 - (0.0158333\ ft)^2))]\} \\
&= 0.24\ F/s \text{ maximum hot spot heat up rate at 1000 F}
\end{aligned}$$

If the core uncovers, the hot spot will heat up and reach temperatures at which metal water reaction becomes significant with respect to the core decay heat. At full power conditions the temperature for significant metal-water reaction is roughly 1800 F, however, it is estimated that metal-water reaction can become significant at 1400 F with the low decay heat power levels. Based on the above calculation, with no credited heat transfer whatsoever, if the hot spot uncovers when the initial temperature is 500 F, it will take roughly 3800 seconds (900 F / 0.24 F/s) to reach approximately 1400 F. This gives the operators more than one hour to restore ECCS flow if it is lost for any reason during the Mode 3 NOP/NOT test.

If a SBLOCA occurs in the RCS piping and insufficient ECCS injection is provided, the core could uncover because of core boiloff. If the vessel level was at the top of the core, it would have to decrease by at least a foot of collapsed liquid to create any potential for significant fuel pin heatup. The cross-sectional area of the core, core bypass, core baffle, and downcomer within the heated core region is roughly 100 ft^2. When the RCS is saturated at 900 psia and 532 F (the nominal maximum temperature during the planned Mode 3 NOP/NOT test), the liquid density is ~47 lbm/ft^3 and heat of vaporization is 670 Btu/lbm. At a core power of 1360 Btu/s, it would take more than 2300 seconds (100 ft^3 * 47 lbm/ft^3 * 670 Btu/lbm / 1360 Btu/s) for each foot to be boiled off. It would take roughly two hours to uncover the high power locations from boiloff.

The adiabatic heatup rate is coupled with the boiloff time from the top of the core would realistically take 2 to 3 hours from the time of core uncovering until the core could reach temperatures of 1400 F for an RCS pipe break. This gives the operators ample time to restore ECCS flow if it is lost for any reason during the Mode 3 NOP/NOT test. The boiloff time is, however, not appropriate if an IMI nozzle break is postulated. For these scenarios, the core liquid level decreases from liquid break flow out of the bottom of the vessel as well as boiloff. The time to achieve the cladding heatup to 1400 F could be much closer to the adiabatic heatup rate because of the break liquid discharge.

4. DB-1 Mode 3 Test Low Decay Heat LOCA Evaluation

Typically a SBLOCA progresses through five phases: (1) subcooled depressurization (2) reactor coolant pump and loop flow coastdown and natural circulation, (3) loop draining, (4) boiling pot, and (5) refill and long-term cooling. The subcooled depressurization phase begins at the leak initiation. The break discharge causes the RCS to depressurize at a rate proportional to the break area. If loss-of-offsite power (LOOP) does not occur, the RCPs remain in operation until there is a loss of subcooling margin and the operator trips them. If LOOP occurs, the RCPs coastdown immediately. AFW is initiated after RCP trip and will control to the natural circulation level.

The proposed DB-1 Mode 3 test primarily uses the energy addition from the four RCPs to heat the RCS fluid. Makeup, letdown, pressurizer sprays and heaters will be used to control the RCS inventory and pressure during the heatup. The operators will control the MFW flow to low level limits to remove the RCP heat addition. For this Mode 3 condition, the control rods will either be fully inserted with trip breakers open, or a single safety group of control rods will be withdrawn to provide a prompt source of trippable negative reactivity, with a sufficient RCS boron concentration to keep the core shutdown. Therefore, the core power will remain essentially constant at the decay heat level of 1.44 MWt (based on the infinite operation with 1.2 times the ANS 1971 fission product decay and B&W heavy isotopes of 101 fuel assemblies).

The initial RCS temperature is specified to be less than or equal to 532 F, with the pressure in the range of 2150 to 2200 psia. For the extremely low core power conditions, the core power is not sufficient to change the bulk RCS liquid temperature. Without any significant heat addition, the RCS will not begin to boil until the RCS pressure drops below the saturation pressure for 532 F, approximately 900 psia. Prior to saturation of the RCS liquid, the pressure will drop below the low RCS pressure SFAS setpoint. Activation of HPI will provide more than enough flow to match the core decay heat. For larger breaks, the RCS will rapidly depressurize and saturate. After subcooling margin is lost, the operators will trip the RCPs if they are still operating. However, for smaller breaks, the HPI mass flow rate can reach an equilibrium with the break mass flow rate. Because of this equilibrium, the RCS loop liquid inventory will remain full and the RCS pressure will remain above the saturation pressure. Adequate subcooling margin may be retained for the smaller break sizes.

The RCS response to a LOCA in Mode 3 is highly break size dependent. Therefore, the responses will be considered for the six break classification ranges used for the full power analyses. The six categories of break sizes and their area ranges used in the full power analyses are:

1. SBLOCAs that may not interrupt natural circulation with MUP 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 0.0035 ft^2).

3. SBLOCAs that allow the RCS pressure to stabilize at approximately the secondary side pressure (0.0035 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).

The descriptions listed above are based on a LOCA occurring while the reactor is at full power. For low power Mode 3 considerations the RCS behavior is different because there is no forcing function for robust natural circulation or for RCS repressurization. Without any core heat generation, the secondary side has no potential to repressurize to the main steam safety valve pressures without RCP energy addition. 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.

4.1. Category 1: 0.002- to 0.005-ft^2 Low Power Mode 3 SBLOCAs

A LOCA is defined as any break that is in excess of the makeup system capacity. The minimum break size is not easily defined because it is dependent on RCS fluid temperature, makeup and letdown flow rates, the critical flow model used in the analysis, and operator actions that are credited. Accordingly, a range of break areas can be given as the minimum break size for a LOCA. For DB-1, a ¾" Schedule 160 (A=0.00206 ft^2) line break is the minimum break size that is classified as a LOCA for full power operation (Reference 7). This minimum break size has been used in the Mode 3 LOCA evaluation as well, even though the criteria used to determine if the makeup system could keep up with the break flow is based on not achieving a reactor trip within 10 minutes. Reactor trip has little meaning for the Mode 3 analyses. Therefore, either a different basis has to be developed or the break size used in the full power work adopted. In this case, the full power basis was selected for use.

The break sizes between 0.00206 to 0.005 ft^2 are in excess of the makeup system flow delivery and will depressurize from the break energy relief to achieve a low RCS pressure SFAS trip within 884 seconds (Reference 6) without any makeup pump flow. On SFAS, the HPI pumps start and letdown is isolated. Break-HPI flow balance calculations (Appendix A) show that full flow from one DB-1 HPI pump will match the saturated break flow rate between 1400 and 1500 psia for Category 1 break sizes (Figure A-2). These pressures are well above the initial RCS saturation pressure of roughly 900 psia (corresponding to 532 F). Therefore, the RCS will remain subcooled, and pressurized. When the break is subcooled, the break flow will be slightly higher and the equilibrium pressure that the RCS achieves can be somewhat lower than that calculated for saturated liquid mass balance. The actual pressure that the system evolves to is not as important as the understanding that the RCS will evolve to a hot, stable condition with the RCS full of liquid that can be maintained as long as the HPI flow matches the leak rate.

If the flow through the break is highly subcooled (i.e. enthalpy roughly 35 to 45 percent of the saturated value) or if the HPI inflow is reduced (as with an HPI line break), the HPI/break flow equilibrium will be at lower pressures. The highly subcooled break flow for break sizes in this category could reach a pressure equilibrium in the 650 to 1400 psia range (Figure A-2). Any pressures below roughly 900 psia would however be saturated. Therefore the expected pressure range for equilibrium for these break sizes is likely in the 1000 to 1400 psia range. The larger break sizes in this Category will depressurize to roughly the saturation pressure at 532 F and cause an early LSCM, while the RCS pressure would hang up with a smaller break size and retain at least 20 F subcooling. Cases with the RCPs tripped and a break location that retains higher subcooling would find a similar equilibrium pressure to the HPI line break because the subcooled conditions increase the break flow rate. Appendix A gives additional information on the break-ECCS flow equilibrium. After the plant reaches the pressure equilibrium, it is in a stable, but hot condition. It will remain there until the operators take action to depressurize the RCS.

Typical LOCA analyses conclude that acceptable long-term core cooling is demonstrated when the RCS remains in hot stable conditions with the HPI pumps remaining operable during the sump recirculation phase. If the HPI pumps are lost during the sump recirculation phase with significant core decay heat, this break range is challenging, because the break is too small to assure depressurization to LPI pressure without operator intervention. Operator-initiated RCS cooldown to DHR has not specifically been analyzed or credited in the existing full power LOCA analyses because it is not needed to demonstrate that a path to long-term cooling is available assuming HPI pump operation during the sump recirculation phase.

If the HPI pumps are not operable during the sump recirculation phase during this Mode 3 test, then there are at least two options for addressing acceptable core cooling. One option is to leave the system at the hot, pressurized conditions and make the BWST to sump switchover. If the HPI pumps continue to operate, then there is no problem and 10 CFR 50.46 criteria will be met. If the pumps should fail, then the RCS will eventually depressurize from the break mass flow, reaching the CFT pressure range and ultimately the upper range for LPI flow. Although this scenario is not recommended (because the operators have little if any control of the RCS evolution), it will still result in acceptable 50.46 consequences.

The second option (recommended for this Mode 3 test) is for the operators to initiate a depressurization to bring the plant down to near LPI flow pressures before the sump switchover occurs. This recommended approach will allow the operators to have some control over the system evolution. Since this is the preferred option, guidance must be developed for the operator-initiated cooldown. In the event that the operators fail to perform the prescribed actions on a timely basis, this scenario could effectively default to the first option. In either case the evaluations performed demonstrate that acceptable core cooling is provided for these small break sizes.

The operators can depressurize the RCS from this stable hot equilibrium condition following a Category 1 LOCA from lower power Mode 3 stable hot conditions by a number of means, depending upon the status of the RCPs and availability of equipment (safety or non-safety). If the RCPs are in operation and the steam generators are available, the forced RCS circulation promotes primary to secondary heat transfer. The operators can use steam generator depressurization to reduce the RCS temperature and throttling of the HPI to manage core exit subcooling margin for pressurized thermal shock (PTS) conditions. This is the preferred depressurization option. If the RCPs are not in operation, the steam generators will be relatively ineffective because there is insufficient decay heat to create effective natural circulation flow rates that can cool the bulk of the RCS fluid. In this scenario, the RCS can be depressurized using feed and bleed cooling through the PORV or other RCS venting paths. If these two methods are unavailable or are not effective for any reason, then the HPI pump flow must be throttled when the RCS is not adequately subcooled. Although this method should be the last resort, it uses only safety-related equipment to complete the depressurization. These methods are presented with greater detail in the following paragraphs.

If the RCPs are still operating and the steam generator can be depressurized, it will induce SG heat removal to decrease the RCS temperature. The reduction in RCS liquid temperature will help maintain subcooling such that the HPI can be throttled based on the PTS guidance. If this method can be established, the RCS can be depressurized in a subcooled state to the LPI pressure range with steam generators and HPI flow throttling. Although this is the normal process, the viability of depressurizing the steam generator is typically not credited in LOCA applications other than through the steam used for the turbine-driven AFW pump(s). Use of the atmospheric vent valves (AVVs) or the turbine bypass valves is not used in the LOCA analyses because they are non-safety systems that are considered unavailable during the event. As discussed previously, since no post-LOCA electrical transfer would be required and no grid disturbance would result, the probability of loss of offsite power would be very low. Additionally, although the air operated AVVs do not have a safety-grade air supply, the AVVs themselves are safety grade and can be manually actuated. Nonetheless, credit for the AVV actuation has not been taken in the existing full power LOCA analyses.

If the RCPs are not in operation, or the steam generator depressurization cannot be accomplished, then a feed-and-bleed cooling method should be initiated. Opening the PORV or other RCS venting paths (PZR sample vent line, letdown, high point vents, etc. with a cumulative effective flow area similar to the PORV area), will initiate a feed-and-bleed cooling method that is very effective in decreasing RCS pressure. Opening the RCS vent pathways will likely result in a loss of subcooling margin (LSCM) and RCP trip if they are still in operation. Although the PORV may be available and is essentially powered, it is not redundant and is typically not credited in the LOCA applications.

The last alternative for depressurizing the RCS is to throttle the HPI pump flow to the extent possible to manage core exit subcooling margin. The amount of throttling depends on the break location and how much ECCS reaches the core. If the break is in the CLPD or an incore measurement instrument (IMI) nozzle, the HPI can be discharged out the break before it reaches the core exit. Loss of ECCS flow through the core will reduce the core exit subcooling and minimize the amount of HPI throttling that can be performed, while still maintaining adequate RCS subcooling. This limitation on the allowable reduction in the HPI flow will keep the RCS pressure well above the LPI flow range for these small break sizes, which would not be desired.

The operators could still depressurize the RCS by additional throttling of the HPI pump(s) flow in a saturated condition. When the HPI flow is throttled, the ECCS injection flow will not match the leak flow. The net loss of inventory will cause the RCS to depressurize and flash, creating steam bubbles that collect in the upper head of the reactor vessel or hot leg U-bend regions. The operators are directed to use hot leg level indications if available to monitor the level decreases. The core-exit thermocouples are also available if the levels fall below the hot legs and possibly into the core region. Therefore, use of all available information is key in preventing this scenario from creating an inadequate core-cooling issue.

Because of the potential core cooling problems, ECCS throttling when adequate subcooling margin does not exist is not allowed by the generic EOP Technical Bases Document (TBD). This is to prevent core uncovering that could evolve through the inadequate core cooling (ICC) regions and into a severe accident. Nonetheless, this EOP guidance for throttling the ECCS is primarily provided for LOCA mitigation when the plant operation in Modes 1 and 2, although it is sound guidance that is normally applicable to the lower modes of operation as well.

The HPI throttling guidance can be modified somewhat for this low power NOP/NOT test because the core boiloff rates are much lower. **The HPI should not be throttled when adequate subcooling margin does not exist with substantial core decay heat power generation rates because the core will uncover and heat up.** Once the core uncovers, the fuel pin temperatures will rapidly increase at the hot spot at rates in the order of 5 F/s (based on the decay heat for infinite operation at one-hour post trip). At this heatup rate, the hot spot can heat up by 1000 F in roughly 200 seconds, which gives the operators little time to respond to the ICC conditions. In contrast, for the low decay heat Mode 3 test, the core could uncover for more than one hour with the fuel pins still remaining below 1400 F (the threshold for significant metal water reaction). Therefore, should the operators throttle the HPI flow back too quickly, they have considerable time to reopen the HPI injection paths before the core conditions could degrade to an unacceptable peak cladding temperature (PCT) or unrecoverable condition. This time is roughly two orders of magnitude longer for the low power Mode 3 conditions than it would be at full power conditions. Therefore, additional ECCS throttling options can be utilized in this unique low power scenario to assist in depressurizing the RCS to LPI flow ranges prior to the BWST to sump switchover.

Framatome ANP performed a representative SBLOCA analysis of an 0.0021-ft^2 IMI nozzle break (Reference 6) to test the HPI throttling method that was proposed with only safety-related equipment. After break opening the RCS depressurized from the break mass and energy release achieving a low RCS pressure reactor trip signal and a low pressure SFAS trip. The rods were already in the core, so the only effect of the reactor trip signal at 286 seconds was a control system setpoint bias application and resultant closure of the turbine bypass line. At 884 seconds, the low pressure SFAS trip was activated. The activation of SFAS initiated one HPI pump and isolated letdown. By

2000 seconds, the RCS had reached a quasi-steady pressure of 1480 psia with the HPI and break mass flows in equilibrium. The pressurizer emptied, but the remainder of the system was liquid full with roughly 45 to 50 F subcooling margin with the RCPs still operating.

The RCS will remain steady at these conditions until some operator action is taken to depressurize the system. Some of the options for depressurizing the system were described above, however, they are also options that are not traditionally credited in the LOCA analyses. The analysis that was performed depressurized the RCS by throttling the HPI, because it is the only safety-related alternative. No credit was taken for RCS relief paths or steam generator depressurization to improve the subcooling margin.

By 2000 seconds, the RCS pressure equilibrium condition had been established in the analysis. Rather than wait hours to initiate the depressurization, the analysis initiated a preprogrammed 1.5-hour linear throttling HPI ramp at 4000 seconds. At 7400 seconds (i.e. less than one hour later), the HPI flow has been throttled to roughly 37 percent full flow, with the RCS pressure at 1260 psia and decreasing. At this time, the SCM dropped to 20 F and the RCPs were tripped. AFW actuated and filled the steam generators to the LSCM level. After RCP trip, the HPI throttling continued at half the initial HPI throttling rate (i.e. 3-hour throttling ramp) that led to HPI termination at roughly 11,400 seconds. The RCS pressure was 880 psia at that time. Between this time and 24,000 seconds the RCS liquid inventories declined from the break mass flow, while the RCS pressure decreased slowly to the CFT fill pressure. The CFTs flowed slowly until 37,000 seconds when they were nearly empty and isolated with the RCS pressure at approximately 220 psia. The minimum reactor vessel mixture level remained above the top of the core. The RCS depressurized after CFT closure achieving LPI flow and refilling the system over the next 2000 seconds. Thereafter the RCS reached a new equilibrium with the core upper plenum pressure at 215 psia. The LPI and break flows were matched at 19 lbm/sec or roughly 140 gpm. The RCS will remain at these conditions indefinitely without reliance on HPI other than to assure redundancy of injection locations if only one LPI line is providing the LPI flow. If the break were 0.005 ft^2, the break and LPI flow at this equilibrium is estimated to be 45 lbm/sec (19 *0.005/0.0021) or 330 gpm. There is ample BWST inventory available to assure that the credited HPI pump flow during the depressurization was based on the clean BWST water supply. Prior to the time of sump switchover, the operators should open the LPI crosstie line and balance flow or provide an alternate injection location such as the LPI auxiliary pressurizer spray (APS) flow in case the break was a crack in the CFT nozzle supplying the LPI flow.

With the operator action to decrease the RCS pressure to the LPI flow range, there is ECCS flow such that the core cooling requirements are met. The only difficulty is that the generic EOPs (Reference 8) specify that HPI cannot be terminated until at least 1000 gpm of LPI flow can be delivered to each CFT nozzle. Achieving at least 2000 gpm of total LPI flow is not possible for these break sizes. Therefore, the HPI termination criteria will not be met. Given that the decay heat is so low for this test, it is reasonable to drop the LPI flow requirement to 500 gpm per line before HPI is terminated. The basis for decreasing the LPI flow requirement is based on the minimal core boiloff contribution for this Mode 3 test. If the LOCA occurred at full power, the

Appendix K decay heat level at ten minutes post trip is equal to a normalized power of 0.02820 (Reference 4). The boiloff rate from 1.02 times 2772 MWt using an enthalpy change of roughly 1100 Btu/lbm ($h_{g\,sat} - h_{BWST}$) is 69 lbm/s [2772 MWt * 1.02 *0.02820 * 948 Btu/s/MWt / (1100 Btu/lbm)]. For a BWST temperature of 90 F, this boiloff mass flow rate is equivalent to roughly 500 gpm [(69 lbm/s * 7.48 g/ft^3 * 60 sec/min) / 62.1 lbm/ft^3]. At this time, if 1000 gpm of ECCS is assured of reaching the core, one half of it is sufficient to remove the core decay heat while the other half initiates refill and ultimately provides an abundance of flow for core cooling. The core boiloff rate is much lower for the current Mode 3 decay heat level. The flow needed to match the core decay heat is roughly 9 gpm [(1360 Btu/s * 7.48 g/ft^3 * 60 sec/min) / (62.1 lbm/ft^3 * 1100 Btu/lbm)]. If the HPI flow is terminated with only 500 gpm of assured LPI flow (500 gpm per LPI line), then the remaining 490 gpm of LPI flow for the Mode 3 test gives roughly the same excess ECCS flow provided by the 1000 gpm for full power LOCA analyses. It should be noted that the excess ECCS for the full power example would increase after 10 minutes as the decay heat decreased with time post trip. Since the decay heat is constant for the Mode 3 test, the additional excess ECCS is not achieved with the 500-gpm flow. The order of magnitude of the excess remains similar, however, and therefore it is concluded that the flow for HPI termination can be decreased to 500 gpm for this low decay heat NOP/NOT test.

The decreased HPI termination criteria will not be met for these smallest LOCAs even if two LPI pumps are available or the LPI cross tie line is open and flows balanced. If two LPI pumps are in operation, there will be adequate ECCS injection to provide the roughly 10 gpm of core cooling needed if the HPI pump(s) are lost. If one LPI is providing flow, the crosstie line should be opened and the flow balanced to provide adequate flow to the core in the event HPI flow is lost. If only one LPI pump is in operation and the LPI crosstie line cannot be opened or the flow balanced, then the LPI auxiliary spray flow should be initiated to provide a redundant flow path for ECCS injection without HPI. Alternatively, if the HPI pump(s) stopped during sump recirculation, but the HPI flow paths to the RCS remain intact without increased leakage from the inoperable HPI pump, some of the LPI piggy-back suction flow to the HPI pump could still flow into the HPI nozzles. Although there may not be much flow, the required flow to match core boiloff is small and the piggy-back flow may be adequate. This path cannot be used if there is excessive leakage from the HPI pump or flow paths when the HPI pump failed.

The RCS pressure for this case without any additional RCS venting paths open will remain above the range at which the core boron concentration needs to be managed to prevent boron precipitation (i.e. greater than 92 psia or a saturation temperature of 322 F – see Figure A-2). In the event that the break size is augmented by PORV or RCS vent paths, then the RCS pressure could stabilize at lower values that may require long-term boron concentration control (the concentration control could be needed after several weeks as discussed in Section 4.4 and given in Table 4-1). If the HPI pump is not available to provide HPI auxiliary pressurizer spray (APS), then the LPI pump can be aligned to the APS line with at least 40 gpm of flow that will provide the boron concentration control if deemed necessary. With a total core decay heat contribution of 1360 Btu/s, roughly 1.3 lbm/sec [$Q_{core}/(h_g - h_{ECCS})$ = 1360/(1100-58)] of flow [i.e. ~10 gpm (1.3 lbm/s / 0.1384 gpm / lbm/s)] is needed to match the core decay heat. Any additional flow would provide a reverse core flow for a cold leg break and provide adequate core boron concentration control.

Although not quantified and credited in these evaluations, the very small LOCAs have the lowest leak rates and will likely produce the least amount of CTMT debris. These break sizes will also have very low ECCS flow rates from the LPI during the sump recirculation phase. The containment spray actuation will not occur for these break sizes, therefore the low ECCS flow rates from CTMT will transport the least amount of foreign material to the CTMT emergency sump. Therefore, HPI pump operation during CTMT sump recirculation, as credited in the existing design basis, might be possible and would be added defense to the evaluations that did not credit it.

4.2. Category 2: 0.005- to 0.035-ft^2 Low Power Mode 3 SBLOCAs

The Category 2 break sizes, between 0.005 to 0.035 ft^2, are well in excess of the makeup system flow capability and will depressurize from the break energy relief to achieve a low RCS pressure SFAS trip within the first 10 minutes after break opening. On SFAS, the HPI pumps start and letdown is isolated. Figure A-2 shows that full flow from one DB-1 HPI pump will match the saturated break flow rate for an 0.01 ft^2 break at roughly 900 psia. Therefore, the smallest Category 2 break sizes should remain at pressures well above the saturation pressure for 532 F. The RCS will remain subcooled and pressurized between 900 and 1300 psia, in a hot, stable condition with the RCS full of liquid. The larger Category 2 breaks will depressurize the RCS and find a saturated equilibrium at pressures between 200 and 900 psia based on the break area (see Figure A-2). These larger break sizes will depressurize and cause an early LSCM (within several minutes) where the smaller sizes could retain more than 20 F subcooling until the operators take action to depressurize the RCS.

After the plant reaches the pressure equilibrium, it is in stable, but hot condition. It will remain there until the operators take some action to depressurize the RCS. The depressurization options available for the operators are identical to those described for the Category 1 methods. If the RCPs are in operation, the operators can use steam generator depressurization to reduce the RCS temperature and throttle HPI to manage core exit subcooling margin to control the pressurized thermal shock (PTS) conditions. This is the preferred depressurization option. If the RCPs are not in operation, the steam generators will be less effective because there is insufficient decay heat to create substantial natural circulation flow rates that can cool the bulk of the RCS fluid. In this scenario, the RCS can be depressurized using feed and bleed cooling through the PORV or other RCS venting paths. If these two methods are unavailable or are not effective for any reason, then the HPI pump flow should be throttled even with the RCS not adequately subcooled. Although this method should be the last resort, it uses only safety-related equipment to complete the depressurization.

Although no analyses were performed for break sizes in the range, the transient evolution is similar to but faster than that of the Category 1 break sizes with only credit for safety-related equipment. After break opening, the RCS depressurization from the break mass and energy release achieves a low RCS pressure reactor trip signal and a low pressure SFAS. After the low pressure SFAS is reached, flow from one HPI pump is established and letdown isolated. The RCS will reach its initial quasi-steady pressure with the HPI and break mass flows in equilibrium. The pressurizer will be empty, but the remainder of the system will be nearly full of liquid. The smallest break sizes in the category should have adequate subcooling margin and the RCPs will still be operating if offsite power was available. The larger break sizes will lose subcooling margin, and the RCPs would be tripped.

If the operator-initiated RCS depressurization with non-safety or non-redundant equipment (described in Section 4.1) is unsuccessful, the operators can still use only safety grade equipment to throttle HPI and reduce RCS pressure in an inadequate subcooled condition. As the operators throttle the HPI pumps the RCS pressure will drop and subcooling margin may be lost if it had not been lost previously. On LSCM the operators will trip the RCPs and AFW will be actuated to fill the steam generators to the LSCM level. After RCP trip the HPI throttling will continue to depressurize the RCS to the CFT fill pressure. The CFTs will flow slowly until they are isolated with the RCS pressure near the LPI injection pressure. Upon CFT isolation to prevent nitrogen from being injected to the RCS, the RCS pressure will be low enough to allow LPI flow to refill the system. Thereafter the RCS will be at a new equilibrium with the core upper plenum pressure between 210 and 150 psia. The LPI and break flows will achieve a matchup point with flows less than 275 lbm/sec or roughly 2000 gpm for these break sizes with the subcooled flows given Table A-3. The RCS could remain at these conditions indefinitely without reliance on HPI other than to assure redundancy of injection locations if only one LPI line is providing the LPI flow. Prior to the time of sump switchover, the operators should open the LPI crosstie line and balance flow or provide an alternate injection location such as the LPI APS flow in case the break was a crack in the CFT nozzle supplying the LPI flow. There is ample BWST inventory available for HPI flow during the operator induced RCS depressurization to LPI pressures that began several hours into the event.

With the operator action to decrease the RCS pressure to the LPI flow range, one of the HPI flow requirements is successfully managed. The only difficulty, as for the smaller breaks, is that the generic EOPs require an HPI pump to be operated if at least 1000 gpm of LPI flow cannot be delivered to each CFT nozzle. Achieving an assured total LPI flow of at least 2000 gpm will not be occur for these break sizes. Therefore, the HPI termination criteria will not be met for the smaller break sizes. Given that the decay heat is so low for this test, the HPI termination criteria can be decreased to a flow requirement of 500 gpm of assured LPI flow per line (See Section 4.1) before HPI pump(s) can be stopped.

The decreased HPI termination criteria will not be met for the smaller Category 2 LOCAs even if two LPI pumps are available or the LPI cross tie line is open and flows balanced. It may be met for some of the larger break sizes if flow is provided to both LPI lines (from two LPI pumps or one pump with the crosstie line open and flows balanced.) If two LPI pumps are in operation and the flow paths open to the RCS, there will be adequate ECCS injection for the smaller break sizes to provide the 10 gpm of core cooling needed if the HPI pump(s) are lost. If one LPI is providing flow, the crosstie line should be opened and the flow balanced to provide adequate flow to the core in the event HPI flow is lost. If only one LPI pump is in operation, and the LPI crosstie line cannot be opened or the flow balanced, then the LPI auxiliary flow should be initiated to provide a redundant flow path for ECCS injection without HPI. Alternatively, if the HPI pump(s) have stopped during sump recirculation, but the HPI flow paths to the RCS remain intact without increased leakage from the inoperable HPI pump, some of the LPI piggy-back suction flow to the HPI pump could still flow into the HPI nozzles. Although there may not be much flow, the required flow to match core boiloff is small and the piggy-back flow may be adequate. This path cannot be used if there is excessive leakage from the HPI pump or flow paths when the HPI pump failed.

The RCS pressure for breaks in this range, without any additional RCS venting paths open, will remain in the range of 150 to 210 psia. These pressures are above the pressure (92 psia) at which the core boron concentration needs to be managed to prevent boron precipitation. In the event that the break size was augmented by PORV or RCS vent paths, then the RCS pressure could stabilize at lower values that may require long-term boron concentration control (the concentration control could be needed after several weeks as discussed in Section 4.4 and given in Table 4-1). If the HPI pump is not available to provide HPI APS, then the LPI pump can be aligned to the APS to provide the boron concentration control if deemed necessary.

4.3. Category 3: 0.035- to 0.06-ft^2 Low Power Mode 3 SBLOCAs.

As the break size increases, the break discharge will result in a LSCM within an estimated two minutes. The RCS pressure will not reach equilibrium early in the event because the flow from one HPI pump cannot match the break flow. Once the RCS saturates, the pressure decrease will slow, but it is estimated that it will drop to the CFT pressure within roughly 10 to 20 minutes. The depressurization will continue until the CFTs are empty and eventually reach the LPI flow range without any operator actions to depressurize the system. The operators should isolate the CFTs as they empty (if possible) to prevent the nitrogen from being injected into the RCS. Once on LPI, the break flow will be matched by the injection rate and the system will become stable at pressures between 120 (maximum area with subcooled conditions from Figure A-3) and 200 (minimum area with saturated conditions) psia. The HPI flow will not be needed after the RCS reaches significant LPI flow except to assure redundancy of injection locations if only one LPI line is providing the LPI flow. Prior to the time of sump switchover, the operators should open the LPI crosstie line and balance flow or provide an alternate injection location such as the LPI APS flow in case the break was a crack in the CFT nozzle supplying the LPI flow. There is ample BWST inventory available for HPI suction during the natural RCS depressurization to LPI pressures.

The generic EOPs allow the HPI pumps to be terminated if at least 1000 gpm of LPI flow can be delivered to each CFT nozzle. Achieving at least 2000 gpm of total LPI flow is unlikely for the smaller Category 3 break sizes. Therefore, the HPI termination criteria may not always be met. Given that the decay heat is so low for this test, the HPI termination criteria may be decreased to a flow requirement of 500 gpm of assured LPI flow per line (See Section 4.1) before HPI pump(s) can be stopped. The HPI termination criteria may be met with this reduced flow rate for all but the smallest break sizes in this category if two LPI pumps are available or the LPI cross tie line is open and flows balanced. If the LPI is only flowing to one line, the LPI APS method or flow through inoperable HPI pump(s) (with minimal leakage) to the HPI nozzles provides redundant ECCS injection locations.

The RCS pressure for this category of breaks should not reach pressures at which the core boron concentration needs to be managed to prevent boron precipitation because the LPI flow causes the pressure to hold-up. In the event that the break size is effectively increased by use of the PORV or other RCS vent sizes, then the RCS pressure could stabilize at lower values that may require long-term boron concentration control. The concentration control could be needed after the time and temperature conditions discussed in Section 4.4 and given in Table 4-1. If the HPI pump was not available to provide HPI APS, then the LPI pump can be aligned to the APS to provide the boron concentration control if necessary.

4.4. Category 4: 0.06- to 0.25-ft^2 Low Power Mode 3 SBLOCAs

The Category 4 range of break sizes (from 0.06 to 0.25 ft^2) will depressurize the RCS to the CFT pressure in roughly 10 minutes or less. The break discharge will result in a LSCM within the first minute for these break sizes. The RCS pressure will likely reach the LPI pressure range within the first 30 minutes of the event without any operator actions to depressurize the system. The operators should isolate the CFTs as they empty (if possible) to prevent the nitrogen from being injected into the RCS. The HPI flow will not be needed after the RCS reaches significant LPI flow except to assure redundancy of injection locations if only one LPI line is providing the LPI flow. Prior to the time of sump switchover, the operators should open the LPI crosstie line and balance flow or provide an alternate injection location such as the LPI APS flow in case the break was a crack in the CFT nozzle supplying the LPI flow. There is ample BWST inventory available for HPI suction during the natural RCS depressurization to LPI pressures.

These breaks depressurize quickly to the LPI flow range. The generic EOPs allow the HPI pumps to be terminated if at least 1000 gpm of LPI flow can be delivered to both CFT nozzles. Achieving at least 2000 gpm of total LPI flow is likely for the larger breaks in this range but unlikely for the smaller break sizes. Therefore, the existing HPI termination criteria may not be met for all the breaks in this classification. Given that the decay heat is so low for this test, the HPI termination criteria may be decreased to a flow requirement of 500 gpm of assured LPI flow per line (See Section 4.1) before HPI pump(s) can be stopped. The revised HPI termination criteria should be met for this reduced flow rate for these break sizes if two LPI pumps are available or the LPI cross tie line is open and flows balanced. If the LPI is only flowing to one line, the LPI APS method or flow through inoperable HPI pump(s) (with minimal leakage) to the HPI nozzles provides redundant ECCS injection locations.

The break flow will be matched by the injection rate and the system could become stable at pressures between 15 psia (maximum area with subcooled conditions from Figure A-3) and 200 psia (minimum area with saturated conditions). The saturated pressure range could be between 200 and 70 psia from Figure A-3. The RCS pressures for some of the larger break sizes in this range can decrease below the pressure at which the core boron concentration needs to be managed to prevent boron precipitation. The current boron precipitation analyses performed for DB-1 at a core power of 102 times 2772 MWt (Reference 1) showed that if the pressure decreased to 14.7 psia, the core could not reach the solubility limit until 4.94 hours or 17800 seconds. The integrated core decay heat at 17500 seconds is 272.52 full power seconds (Reference 4). The core decay heat rate for the burned fuel is relatively constant at a normalized value of 0.00089. It will take at least 306000 seconds (272.52/0.00089) or 3.5 days to integrate to the same power for the first 4.94 hours of the full power analysis. This time does not consider that the 76 assemblies are fresh fuel without any decay heat (Reference 2). Therefore, decay heat used in the time calculation can be reduced by the ratio of 101/177. The time to reach the solubility limit shifts to 6.2 days with this adjustment.

The time that the core could reach the solubility limits for the full power analyses is a function of RCS saturation temperature. The time for the low decay heat conditions to reach the solubility limit for an RCS temperature of 212 F was determined based on an adjustment to full power calculation. The time to reach the solubility limits for other temperatures were calculated with a similar adjustment and they are given in Table 4-1. The calculations performed in that table use the RCS saturation pressure and temperature listed from Table 8 of Reference 1 with the time to reach the solubility limit. The solubility limit time is converted to seconds and it is the search variable used to get the integrated power from Reference 4 at the time equal to or conservatively less than the solubility time. The integrated power is divided by the whole core decay heat value, 0.000508 (0.00089*101/177) considering that 76 fresh assemblies have been inserted into the core. The time to reach the solubility limit increases with temperature and maximizes at 144 days or 20.5 weeks for a temperature of 320 F (saturation at 90 psia). The operator initiation of the LPI core boron concentration should be performed when the RCS is saturated, but before the temperature decreases below the core exit temperature (considering instrument error) that precipitation could occur. In Reference 1, the uncertainty of 11 F was applied to the core exit temperatures and the saturation temperature plus this uncertainty is given in the last column of Table 4-1. The HPI or LPI APS methods can provide the required flow for boron concentration control.

4.5. Category 5: 0.25- to 0.75-ft^2 Low Power Mode 3 SBLOCAs

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. These Category 5 break sizes will depressurize the RCS to below 100 psia within 10 minutes if only one LPI train is in operation. The operators should isolate the CFTs as they empty (if possible) to prevent the nitrogen from being injected into the RCS.

These breaks depressurize quickly to the LPI flow range. The generic EOPs allow the HPI pumps to be terminated if at least 1000 gpm of LPI flow can be delivered to both CFT nozzles. Achieving at least 2000 gpm of total LPI flow would be assured for these break sizes. Therefore, the HPI termination criteria should be met if two LPI pumps are available or the LPI cross tie line is open and flows balanced. If the LPI is only flowing to one line, the LPI APS method or flow through inoperable HPI pump(s) (with minimal leakage) to the HPI nozzles provides redundant ECCS injection locations.

The RCS pressure for this entire category of breaks will reach RCS pressures at which the core boron concentration needs to be managed to prevent boron precipitation. Table 4-1 provides the time for initiation of the post-LOCA boron concentration control method. If the HPI pump is not available to provide HPI APS, then the LPI pump can be aligned to the APS to provide the boron concentration control if necessary.

4.6. 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. Although not considered as a separate category, the LBLOCA spectrum is divided into two break ranges, 0.75 to 2.0 ft^2 and greater than 2.0 ft^2, for the purpose of EM methods (Reference 5). The smaller range is referred to as the transition LOCA range, while the upper range is the traditional LOCA range that contains the typically limiting double-ended cold leg break sizes. All of the Category 6 LOCAs will depressurize to the containment pressure within minutes.

These breaks depressurize quickly to the LPI flow range. The generic EOPs allow the HPI pumps to be terminated if at least 1000 gpm of LPI flow can be delivered to both CFT nozzles. Achieving at least 2000 gpm of total LPI flow is likely for these break sizes. Therefore, the HPI termination criteria should be met if two LPI pumps are available or the LPI cross tie line is open and flows balanced. If the LPI is only flowing to one line, the LPI APS method or flow through inoperable HPI pump(s) (with minimal leakage) to the HPI nozzles provides redundant ECCS injection locations.

The RCS pressure for this category of breaks will reach RCS pressures at which the core boron concentration needs to be managed to prevent boron precipitation. Table 4-1 provides the time for initiation of the post-LOCA boron concentration control method. The LPI pump can be aligned to the auxiliary pressurizer spray line to provide the boron concentration control if necessary.

Table 4-1. Low Decay Heat Mode 3 Boron Concentration Control Times

Actual RCS Pres. (psia)	Actual RCS Temp. (F)	Time to Reach Solubility Limit (hr)	Time to Reach Solubility Limit (sec)	Integrated Decay Heat (Full-Power-Seconds)	Time to Initiate Boron Concentration Control (days)	Uncertainty Adjusted Core Exit Temperature, F
14.7	212	4.94	17784	272.52	6.2	223
20	228	7	25200	353.45	8.1	239
25	240	10.72	38592	477.31	10.9	251
30	250	16.14	58104	643.51	14.7	261
40	267	33.89	122004	1079.85	24.6	278
50	281	49.17	177012	1418.96	32.3	292
60	293	68.89	248004	1781.73	40.6	304
70	303	100	360000	2327.74	53.0	314
80	312	168.89	608004	3338.43	76.1	323
85	316	263.61	948996	4588.05	104.6	327
90	320	419.44	1509984	6321.39	144.1	331

5. Key LOCA Operator Actions to Assure Continued Core Cooling

In Section 3 the HPI functional requirements were identified. In Section 4, the entire LOCA spectrum occurring from the low power Mode 3 conditions was reviewed to determine the RCS system responses. The discussion for each break classification was focused on the potential HPI flow requirements during sump recirculation. For Category 1 and 2 breaks, the challenges noted were achieving the HPI termination criteria of 1000 gpm per LPI line for all break sizes and post-LOCA boron precipitation when the RCS is saturated and the temperature is less than 322 F real (333 F uncertainty adjusted). The discussions provided ways to deal with the HPI flow requirements, although they may require operator action and procedural changes to ensure the HPI pumps are not critical for core cooling during the sump recirculation phase. The modification to the guidance is discussed in detail in this section.

For the break sizes less than 0.035 ft^2 (Category 1 and 2 breaks), the RCS pressure could stabilize at a pressure above the LPI injection range with the HPI flow matching the break flow. Operator actions that will depressurize the RCS include the following and are in order of preference:

1. Depressurize the steam generator(s) to subcool the RCS fluid and throttle HPI as instructed by the EOPs to manage core exit subcooling margin. This method is easily established when the RCPs are in operation and may be successful without RCPs depending upon the amount of RCS natural circulation flow. The steam generator heat removal and RCS heat losses will result in some circulation of fluid within the RCS without RCP operation.

2. Use the PORV (and any RCS venting paths) to depressurize the RCS. These will likely result in LSCM and RCP trip. The additional venting area adds to the break area to give an effective area that could depressurize the RCS below the LPI deadhead pressure.

3. If the first two options cannot be completed or are ineffective, then and only then would the operators be directed to throttle the HPI flow to depressurize the RCS to the LPI pressure range, i.e., less than 215 psia. This guidance would only apply to this low decay heat test and revised EOP actions must be developed. **Throttling HPI when the RCS is saturated is considered only as a last option, but it is one that utilizes only safety grade equipment and instrumentation.**

Use of Options 1 and 2 are typical transient evolutions supported by current EOP guidance. They will depressurize the RCS to the LPI injection pressure while preserving adequate core cooling with non-safety or non-redundant methods. The HPI throttling option is the only option that relies exclusively on safety related equipment. However, this action is in direct contradiction to the current EOPs and it is acceptable only for very low decay heat scenarios with alternate EOP guidance developed specifically for use with this Mode 3 test.

5.1. HPI Throttling Option Revised EOP Guidance

In LOCA scenarios where the RCS pressure has found an equilibrium above the LPI flow pressure, and all other means of depressurizing the RCS have failed, throttling the HPI pump(s) can be an effective means for reducing the RCS pressure. However, the throttling may lead to a loss of subcooling margin at the core exit. **Throttling of the HPI when the core exit fluid is saturated is not supported by the generic EOP guidance and is not recommended under any circumstance with typical post-LOCA core decay heat rates.** The high core decay heat would rapidly lead to core uncovering and clad heatup. This Mode 3 test is unique, however, in that the core has substantially lower core heat because of the long period following its last operation. Proceduralized HPI throttling under saturated conditions for this test can be achieved while preserving acceptable core cooling.

For this Mode 3 low power test, the only reason that a saturated or subcooled RCS will be at elevated pressures following a SBLOCA is with reverse SG heat transfer or because the ECCS flow matches the break flow. There is no plausible reason for reverse heat transfer from the SGs. Therefore ECCS injection from HPI or MU pumps is the cause of the high RCS pressure. Reducing the ECCS flow in an orderly and controlled manner will decrease the RCS pressure. The reduced ECCS flow will absorb some core decay heat or other stored energy in the RCS before being discharged out of the break. The core power is 1360 Btu/s. Approximately 1.3 lbm/s of ECCS flow or roughly 10 gpm is sufficient to remove this core power. Throttling the HPI flow will depressurize the RCS and continue to match or exceed the core decay heat for this special Mode 3 test until the HPI flow is less than 10 gpm to both HPI nozzles if it reaches the core. The only scenario that could seriously challenge continuous core cooling is a break location that could prevent the ECCS flow from reaching the core. A CLPD, HPI line break, or IMI nozzle break each has the potential to bypass ECCS before it reaches the core. The cold leg breaks can discharge the flow from only one of the two HPI injection leg flows. The IMI nozzle break, however, could discharge all the ECCS before it reaches the core.

The general ECCS throttling guidance is acceptable based on current EOP guidance if adequate core exit subcooling exists. If adequate core exit subcooling is lost the RCPs should be tripped. Initiation of the HPI flow throttling method without adequate SCM cannot begin until after the RCPs are tripped. This removes the RCP heat addition plus ensures that the RCS void fraction is not growing to unstable conditions where inadequate core cooling would occur if the RCPs were lost. ECCS throttling should not be initiated unless all preferred RCS depressurization methods could not be used. In this low decay heat test, with the RCS pressure elevated, the operators should begin to slowly throttle the ECCS two hours after break initiation. The throttling should begin with the MU pump(s) if they are in operation at 2 hours after break initiation. Over the next hour the MUP(s) should be terminated. If the RCS pressure is still elevated, throttling the HPI pump(s) can begin. If two HPI pumps are in operation, one pump should be throttled back over the next hour to nothing. By four hours after the break opening, the operators could use this guidance to throttle the available ECCS to a single

HPI pump. If there was only one HPI pump operating, then the operators should not throttle that HPI pump until the four hour point. Then and only then, if the RCS pressure remains elevated, can the throttling of the final HPI pump begin. It should be throttled back slowly while keeping roughly the same flow going to both cold leg legs that are being fed. Over the next six hours, the operators should gradually throttle the HPI flow back to reduce the RCS pressure, while observing the hot leg level indications and core exit thermocouple responses. With this throttling, the RCS should be reduced to the LPI pressure range before the BWST has reached the level that sump transfer was required.

The range of possible break locations and single failures were examined to identify those that seriously challenge the ability to maintain adequate core cooling. Break locations in the hot leg, cold leg pump suction (CLPS), cold leg pump discharge (CLPD), HPI line break, CFT line break, and an IMI nozzle will be considered. Break sizes that depressurize to the LPI pressure range without operator cooldown will not need HPI throttling. They simply need a redundant flow path to assure adequate core cooling if the LPI is flowing to only one line. The redundant flow path is for CFT line break scenarios. RCS break sizes that remain at pressure are the focus of the specific break locations that are evaluated below to show HPI throttling is acceptable.

Hot Leg Break

For a hot leg break, all the HPI or MU pump(s) flow that is injected flows into the reactor vessel through the core before exiting through the break. If the RCS pressure remains elevated, and all other depressurization methods were unsuccessful, then the operators should begin to throttle any ECCS flow as described previously, but reiterated for clarity here. The throttling first begins with any ECCS in excess of a flow from a single HPI pump and should include RCP trip at any time in the process if adequate SCM does not exist. The operators should slowly throttle the MU pump(s) beginning at 2 hours after break initiation. Over the next hour the MUP(s) should be terminated. If the RCS pressure is still elevated, throttling the second HPI pump can begin by hour three and be complete in one hour. If only one HPI pump is in operation, then the HPI pump throttling can wait until the fourth hour. By four hours, there will be only one HPI pump operating with this special EOP guidance. If the RCS pressure is still elevated, then throttling of the final HPI pump should begin. It should be throttled back while providing flow to two HPI nozzles over the next 6 hours. During this throttling, adequate core cooling is provided while the total HPI pump flow is greater than the 10 gpm core boiloff, plus passive metal boiloff and flashing mass loss during depressurizations. Adequate core cooling will be provided for the hot leg breaks as the RCS is slowly but steadily depressurized to the LPI pressure flow range. Adequate core cooling is assured once LPI flow is established. The operators will, however, need to establish a redundant LPI injection path to accommodate a hypothetical CFT line break.

CLPS Break

A CLPS break location will not result in any direct bypass of the pumped injection. Therefore, this break location will be similar in nature to the hot leg break, although it may depressurize at a slightly different rate because the fluid temperatures upstream of the break could be different. The general MU and HPI throttling guidance applies to the CLPS breaks also. Adequate core cooling is provided for the CLPS break while the RCS is slowly but steadily depressurized to the LPI pressure flow range. Adequate core cooling is assured once LPI flow is established. The operators will, however, need to establish a redundant LPI injection path to accommodate a hypothetical CFT line break.

CLPD or HPI Line Break

A CLPD (or an HPI line break, in which the operators balance the HPI flows to two cold legs) can result in roughly one half of the injected HPI flow not reaching the core if the break is in the loop receiving HPI flow. Throttling of the MU pump(s) and HPI pumps for this break location will follow the general method developed for this Mode 3 test. Adequate core heat removal is provided so long as both legs have sufficient flow to match the 10 gpm core boiloff, plus passive metal boiloff and flashing mass loss during slow depressurization. Adequate core cooling is assured once LPI flow is established. The operators will, however, need to establish a redundant LPI injection path to accommodate a hypothetical CFT line break.

CFT Line Break

If the break is on the RCS side of any check valves in the CFT line (including limited break areas in the CFT line) all the HPI injected reaches the core, therefore the general HPI throttling described is appropriate for this break location. Adequate core heat removal is provided so long as both legs have more than 10 gpm core boiloff, plus passive metal boiloff and flashing mass loss during the slow depressurization. Once LPI flow begins, if only one LPI pump is providing flow to only one LPI line, then a separate redundant ECCS flow of more than 10 gpm is needed. Adequate core cooling is assured when flow is provided to the second LPI injection line or alternate location with flow in excess of the 10 gpm.

IMI Nozzle Break

If the break is in an IMI nozzle, then the break areas are limited between 0.0021 and 0.006 ft^2. Throttling for this break location should be initiated as with the other cases although this break location flow can bypass the HPI, LPI, or CFT flow before it enters the core. In this worst case break location, throttling of the HPI will result in RCS inventory loss because the break flow will exceed the ECCS flow. Adequate core cooling was demonstrated for a limiting 0.0021- ft^2 IMI break size with a 1.5 to 2 hour HPI throttle ramp. The general HPI throttle ramp is longer and should be bounded by the shorter ramp. Adequate core heat removal is provided by the RCS inventory so long as the core mixture level remains above the top of the core. Adequate core cooling is assured once LPI flow is established. The operators will, however, need to establish a redundant LPI injection path to accommodate a hypothetical CFT line break.

5.2. Operator Management of the BWST Inventory Depletion

Break sizes that depressurize to the LPI pressure range without operator cooldown will reach these lower pressures well before the BWST empties, with or without building spray actuation. Smaller breaks that tend to hang up in pressure and rely on HPI throttling to slowly reduce pressure will not actuate containment spray, although they will take a substantial amount of time to depressurize the system. Estimates of the BWST inventory depletion are provided for these scenarios. The major difficulty with the estimation of BWST inventory usage is determining the time-averaged RCS pressure and number of MU or HPI pumps in operation. These will be considered based on the general ECCS throttling guidance provided in Section 5.1.

It is assumed that the BWST has a minimum usable volume of 360,000 gallons. If two MU pumps (throttled to give 250 gpm per pump) are in operation for two hours with a linear throttling to zero over the next hour, then they will use 75,000 gallons (2 pumps * 250 gpm/ pump * 2.5 hrs * 60 min/ hr).

An HPI pump will provide roughly 569 gpm at 900 psia. If the RCS pressure remains above this pressure after the makeup pumps are terminated then the HPI could have provided some flow over the 3 hours, although the RCS pressure at which the flow was provided would have been much higher if makeup pumps were in operation. If the average flow rate was 400 gpm, then the total HPI flow from one pump would have been 72,000 gallons (1 pumps * 400 gpm/ pump * 3 hrs * 60 min/ hr). If two pumps were in operation, then the RCS pressure would have been higher and the net HPI flow from both would have been less. If the average flow for the first 3 hours from both pumps was 500 gpm total, then 90,000 gallons (500 gpm/ pump * 3 hrs * 60 min/ hr) of BWST inventory would have been used over the first 3 hours of the event.

If two HPI pumps were in operation at 3 hours, one HPI pump can be throttled back over the next hour. If the average flow for this hour from the throttled pump was 300 gpm total, then 18,000 gallons (1 pumps * 300 gpm/ pump * 1 hrs * 60 min/ hr) of BWST inventory would have been used. The other HPI pump would be flowing at a maximum rate less than 569 gpm for a scenario that remains above 900 psia with one HPI pump providing flow. A flow of 569 gpm for one hour removes an additional 34,000 gallons (569 gpm * 60 minutes).

For the scenario discussed, the BWST inventory at 4 hours could be reduced by 217,000 gallons (75,000 + 90,000 + 18,000 + 34,000). That leaves a minimum of 143,000 gallons (360,000 – 217,000) for the remaining pump before the BWST switchover must occur. If a linear throttling ramp is used with a maximum unthrottled HPI flow of 775 gpm (corresponding to ~200 psia system pressure), then the maximum HPI ramp time is calculated to be 369 minutes (143,000 / 387.5 gpm) or roughly 6 hours.

This BWST draindown example supports the operator actions specified as the recommended throttling options. If the MU pump(s) are in operation, the operators will begin to secure them at two hours, and have them off at three hours. Once the MUPs are secured, the operators will begin to throttle one of the two HPI pumps at three hours, and have it secured by in four hours. At four hours, the operator will begin to throttle the remaining HPI pump, and its throttling over the next six hours will depressurize the RCS to the LPI injection pressure. This guidance will support the HPI pump operation during the prescribed throttling with the minimum usable BWST volume. If the makeup pumps and second HPI are not in operation, then there would be additional BWST inventory remaining at ten hours, assuming that the single HPI pump throttling began at 4 hours.

5.3. HPI Termination Criterion and Alternate ECCS Injection Alignments

Once the RCS has been depressurized to the point where LPI delivery can occur, the ability to terminate HPI can be considered. The current criterion for termination of HPI is that unless the minimum LPI flow of 1000 gpm is provided to both CFT nozzles, HPI is not to be terminated. This minimum flow rate is based upon assuring continued core cooling in the LOCA analyses performed for full power operation and Appendix K decay heat levels based upon infinite operation at full power. **Because of the low core power/decay heat level in this test, reduction of the minimum LPI flow to 500 gpm is recommended.** This flow is more than sufficient to provide continued core cooling. If this reduced flow criterion cannot be met, the HPI pumps should remain in operation. Should they fail because of debris accumulation, the LPI pumps could still provide flow to the RCS through an inoperable HPI pump, assuming that there is nothing more than normal leakage. The amount of flow through an inoperable HPI pump has not been analyzed.

Another alternative exists in the form of redundant ECCS injection via the LPI APS method. **The LPI APS method will provide 40 gpm or more to the RCS at 14.7 psia and this is more than sufficient to match core decay heat.** It will not be sufficient at higher pressures because the flow will be lower. Note: It will take approximately 4 to 5 hours for the APS flow at this flow rate to fill the pressurizer and begin to spill into the core. Slightly higher flow rates can be achieved by throttling the LPI injection valves to increase the LPI pump discharge head.

5.4. Post-LOCA Boron Precipitation Control

Performance of this Mode 3 test is acceptable without a complete analysis of the post-LOCA boron concentration control, because DB-1 has been shut down without any power generation since February 16, 2002. Since that time, 76 fresh (Reference 2), unirradiated fuel assemblies have been loaded with 101 previously irradiated assemblies. The core decay heat is quite low, and the control rods will either be fully inserted with trip breakers open, or a single safety group of control rods will be withdrawn to provide a prompt source of trippable negative reactivity during this Mode 3 test. The decay heat level at this time using Appendix K assumptions is estimated to be a normalized power fraction of 0.00089, which is more than 100 times less than the typical post trip decay heat load. (Using realistic decay heat levels, the estimated normalized power fraction is 0.00021.) Therefore, the decay heat is so minimal that the combination of core inlet subcooling and heat losses through the core barrel wall and other heat structures should be sufficient to prevent the core concentration from exceeding the solubility limit.

If heat losses are excluded, a conservative calculation has been made to show when the core will reach the solubility limit. Because there is no direct measurement of the core boron concentration, it may be prudent to initiate an active method. If the core has lost subcooling margin and is at a pressure that typically requires boron concentration control (i.e. less than 92 psia), sufficient time (greater than 6 days) exists to establish an appropriate boron dilution mechanism if one is deemed necessary. However, current procedures require initiation of an active core boron concentration control method shortly after sump switchover to support use of all active methods. The HPI pump is the primary post-LOCA boron dilution path with flow supplied to the auxiliary pressurizer spray line to initiate a reverse core flow for a CLPD break. If this preferred method is unavailable, the LPI pump can provide adequate flow via the APS method to control the post-LOCA core boron concentration for CLPD breaks with the low decay heat level. Although not explicitly measured or calculated, it is estimated that there would be at least 20 gpm at 100 psia, which would match the core decay heat rate with roughly 10 gpm and retain an additional 10 gpm to provide successful post-LOCA boron dilution.

6. Recommendations and Conclusions

The low decay heat Mode 3 LOCA evaluations discussed in this document demonstrate adequate core cooling scenarios with the special procedural guidance provided in Section 5. This core cooling is demonstrated without HPI pumps during the sump recirculation phase based on the RCS depressurization, backup redundant ECCS flow paths, and a LPI APS flow for boron concentration control. Since adequate core cooling has been demonstrated, the acceptance criteria of 10CFR50.46 will be met. Therefore, it is concluded that it is acceptable to proceed with the Mode 3 NOP/NOT test with the current HPI pumps.

7. References

1. Framatome ANP Document 86-5006059-00, "Post-LOCA Boron Concentration Management for DB," 3/15/2000.
2. Framatome ANP Document 86-5019560-01, "DB FFCD Cyc 14 with New Head," 1/27/03.
3. Framatome ANP Document 32-1159704-00, "Eval EFW Turbine SS-SLB", 2/18/86 FANP Proprietary"
4. Framatome ANP Document 32-1258134-00, "Decay Heat for LOCA Analysis", 9/26/96. FANP Proprietary.
5. Framatome ANP Topical Report, "BWNT LOCA Evaluation Model for OTSG Plants," BAW-10192P-A, Rev. 0, June 1998.
6. Framatome ANP Document 32-5026193-00, "DB-1 Analyses of SBLOCA at Hot Zero Power", May, 2003. FANP Proprietary"
7. Framatome ANP Document 86-5006232-01, "DB-1 LOCA Summary Report," 9/17/2002.
8. Framatome ANP Document 77-1152414-09, "Emergency Operating Procedures Technical Bases Document", 4/19/2000.
9. Framatome ANP Topical Report, "RELAP5/MOD2-B&WLOCA Evaluation Model for OTSG Plants," BAW-10164PA, Rev. 4, November 2002.
10. Framatome ANP Document 86-5004842-00, "DB CY13 FFCD OFF 620 CY12 EFPD," 8/17/1999.

APPENDIX A. Break-ECCS Flow Equilibruim Calculations

If a SBLOCA occurs during the Mode 3 NOP/NOT test at DB-1, the RCS will naturally evolve to quasi-steady equilibrium condition where at some pressure the ECCS flow will match the break flow. The system pressure at which this matchup occurs is based on the ECCS flow and the break size and location. For the ECCS flow, flow from a single injection train was used in the evaluations based on the single HPI pump and single LPI flows in Reference 7. The break location and transient evolution are both important in determining the inlet conditions used in the break mass flow rate. These could result in vastly different break flow rates and RCS pressure at which the break size equilibrates. Rather than attempt to accurately characterize the differences and pick one location or inlet set of conditions several different break inlet conditions or locations will be used. They will range from saturated liquid Moody break discharge to highly subcooled liquid flow rates with Extended-Henry Fauske break discharge.

Table A-1 was developed to calculate the total ECCS flow during the BWST draining period. In Table A-1, the ECCS flows from 1 HPI pump (Table 3-8 of Reference 7) was used to develop a linear and 6th order polynomial curve fit to the data. This curve fit uses a linear fit for pressures less than 800 psia

$$Q_{HPI\ in\ gpm}|_{P\leq 800} = -0.2728 * P_{RCS\ in\ psia} + 829.4$$

and the sixth order polynomial for higher pressures.

$$Q_{HPI\ in\ gpm}|_{P>800}) = \text{MAX}\{0, [-3.210933E\text{-}15 * (P_{RCS\ in\ psia})^6 + 1.401555E\text{-}11 * (P_{RCS\ in\ psia})^5 - 2.30188E\text{-}8 * (P_{RCS\ in\ psia})^4 + 0.00001740185 * (P_{RCS\ in\ psia})^3 - 0.005958842 * (P_{RCS\ in\ psia})^2 + 0.47122 * (P_{RCS\ in\ psia}) + 818.9652]\}$$

This curve fit is shown in Figure A-1 along with the data from Table 3-8 with the 3.5 percent head reduction. The fit is used to calculate the HPI flows at any pressure during the BWST draining period for this evaluation. The 1 LPI pump (Table 3-5 of Reference 7) flows are given in the third column with linearly interpolated values with pressure provided for the pressures not in the table. The sum of the 1 HPI pump fit and 1 LPI pump flow is combined for a total volumetric flow in Column 4. These volumetric flows are converted to mass flow for a BWST temperature of 90 F by multiplying by 0.1384 lbm/s/gpm [62.1 lbm/ft^3 / (7.48 gal/ft^3 * 60 s/min)].

Table A-1. Combined ECCS Flows from BWST for 1 HPI Pump and 1 LPI Pump

1	2	3	4	5
RCS Pressure, (psia)	Total HPI Flow from 1 Pump, (gpm)	Total LPI Flow from 1 Pump, (gpm)	Total CLPD HPI/LPI Flow, (gpm)	Total CLPD HPI/LPI Mass Flow, (lbm/s)
15	825	3307	4132	572
35	820	3307	4127	571
50	816	3161 *	3977	550
95	803	2722	3525	488
100	802	2663 *	3465	480
135	793	2250	3043	421
195	776	1102	1878	260
200	775	827 *	1602	222
215	771	0	771	107
300	748	0	748	103
400	720	0	720	100
500	693	0	693	96
600	666	0	666	92
700	638	0	638	88
800	614	0	614	85
900	569	0	569	79
1000	519	0	519	72
1100	471	0	471	65
1200	430	0	430	59
1300	389	0	389	54
1350	362	0	362	50
1400	323	0	323	45
1450	264	0	264	37
1500	173	0	173	24
1525	111	0	111	15
1550	36	0	36	5

* Linearly Interpolated

Table A-3 shows the development necessary to calculate the saturated liquid break-ECCS mass flow equilibrium point versus RCS pressure.

Table A-2 was developed to calculate the saturated liquid break-ECCS mass flow equilibrium point versus RCS pressure. In Table A-2, Column 2 gives the Moody Critical Mass Flux (Table C.1 Reference 9) versus pressure or linear interpolation with pressure. The corresponding saturated liquid enthalpy for that pressure point is listed in Column 3. The ECCS mass flows from Column 5 of Table A-1 are listed in Column 4 of Table A-2. These flows are divided by the Moody Critical Flow Mass Flux from Column 2 of Table A-2 to give an effective break area (see Column 6) at which the saturated leak mass flow matches the ECCS flow. The ECCS injection flow is supplied at a temperature of 90 F from the BWST. The energy added to the RCS from the ECCS injection is calculated by multiplying the ECCS flow times the injection enthalpy of 58 Btu/lbm. The saturated liquid energy relief for this effective break flow area is calculated by multiplying the critical mass flux times the break area times the saturated liquid enthalpy. The saturated liquid energy relief is given in the last column of Table A-2.

Table A-2. Saturated Liquid Break Area for Leak-ECCS Mass Balance

1	2	3	4	5	6	7
RCS Pres, (psia)	Moody Critical Saturated Liquid Mass Flux, G_f, (lbm/ft^2- s)	Saturated Liquid Enthalpy, H_f, (BTU/lbm)	Total Combined CLPD ECCS Mass Flow, (lbm/s)	Total ECCS Energy Addition to the RCS (Btu/s)	Saturated Liquid Break Area to Match ECCS Flow at RCS Pressure, (ft^2)	Sat. Liquid Energy Flow from Area that Matches ECCS RCS Pressure, (BTU/s)
15	768	180	572	33171	0.7447	103046
35	1351	220	571	33127	0.4229	125767
50	1788	250	550	31922	0.3079	137713
95	2470	294	488	28300	0.1975	143307
100	2546	299	480	27815	0.1884	143171
135	2918	318	421	24423	0.1443	134108
195	3555	353	260	15077	0.0731	91671
200	3608	356	222	12858	0.0614	78814
215	3719	361	107	6187	0.0287	38472
300	4346	390	103	6001	0.0238	40333
400	5085	424	100	5782	0.0196	42284
500	5638	448	96	5563	0.0170	42962
600	6192	472	92	5344	0.0149	43460
700	6648	491	88	5125	0.0133	43363
800	7103	510	85	4932	0.0120	43352
900	7493	526	79	4570	0.0105	41461
1000	7884	543	72	4166	0.0091	38972
1100	8225	557	65	3781	0.0079	36319
1200	8567	572	59	3449	0.0069	34007
1300	8869	585	54	3124	0.0061	31527
1350	9020	592	50	2907	0.0056	29672
1400	9171	599	45	2594	0.0049	26782
1450	9305	605	37	2118	0.0039	22108
1500	9440	612	24	1389	0.0025	14647
1525	9507	615	15	893	0.0016	9466
1550	9574	618	5	287	0.0005	3058

Table A-3 shows the development necessary to calculate the highly subcooled liquid break-ECCS mass flow equilibrium point versus RCS pressure. In Table A-3, Column 2 gives the subcooled Extended Henry-Fauske critical mass flux (Table C.2 Reference 9) versus pressure or linear interpolation with pressure. The highest subcooled liquid enthalpy for each pressure is listed in Column 3. The ECCS mass flows from Column 5 of Table A-1 are listed in Column 4 of Table A-3. The flow at each pressure is divided by the Extended Henry-Fauske critical mass flux from Column 2 of Table A-3 to give an effective break area (see Column 6) at which the subcooled leak mass flow matches the ECCS flow. The ECCS injection flow is supplied at a temperature of 90 F from the BWST. The energy added to the RCS from the ECCS injection is calculated by multiplying the ECCS flow times the injection enthalpy of 58 Btu/lbm. The subcooled liquid energy relief for this effective break flow area is calculated by multiplying the critical mass flux times the break area times the subcooled liquid enthalpy. The subcooled liquid energy relief is given in the last column of Table A-3.

Table A-3. Subcooled Liquid Break Area for Leak-ECCS Mass Balance

1	2	3	4	5	6	7
RCS Pres, (psia)	Extended Henry-Fauske Critical Flow of Subcooled Liq, Mass Flux, G_f, (lbm/ft^2- s)	Subcooled Liquid Enthalpy, H_f, (BTU/lbm)	Total Combined CLPD ECCS Mass Flow, (lbm/s)	Total ECCS Energy Addition to the RCS (Btu/s)	Subcooled Liquid Break Area to Match ECCS Flow at RCS Pressure, (ft^2)	Sub. Liquid Energy Flow from Area that Matches ECCS RCS Pressure, (BTU/s)
15	2293	65	572	33171	0.2494	36963
35	3939	90	571	33127	0.1450	51257
50	5174	109	550	31922	0.1064	59758
95	7100	129	488	28300	0.0687	63127
100	7314	132	480	27815	0.0656	63155
135	8365	141	421	24423	0.0503	59538
195	10166	158	260	15077	0.0256	41075
200	10316	159	222	12858	0.0215	35338
215	10632	162	107	6187	0.0100	17259
300	12423	175	103	6001	0.0083	18147
400	14530	191	100	5782	0.0069	19079
500	16125	202	96	5563	0.0059	19383
600	17719	213	92	5344	0.0052	19606
700	19058	221	88	5125	0.0046	19536
800	20397	229	85	4932	0.0042	19507
900	21574	236	79	4570	0.0037	18620
1000	22751	243	72	4166	0.0032	17469
1100	23808	249	65	3781	0.0027	16237
1200	24864	255	59	3449	0.0024	15165
1300	25829	260	54	3124	0.0021	14020
1350	26311	263	50	2907	0.0019	13178
1400	26793	266	45	2594	0.0017	11879
1450	27240	268	37	2118	0.0013	9788
1500	27688	270	24	1389	0.0009	6474
1525	27911	272	15	893	0.0006	4181
1550	28135	273	5	287	0.0002	1350

Figure A-1. One HPI Pump Head Flow Curve (with 3.5% Head Degradation) Versus HPI Head Flow Curve Fit

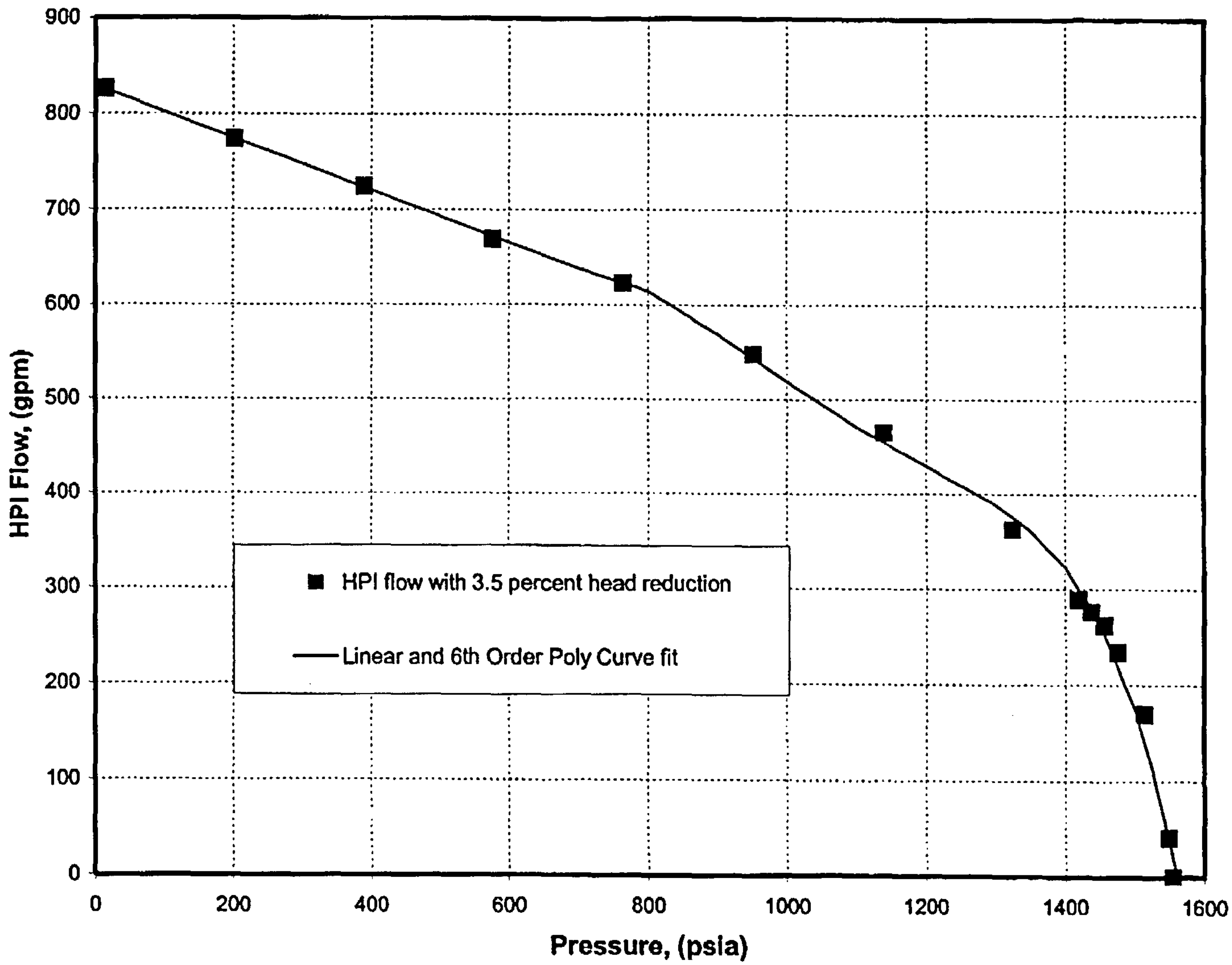


Figure A-2. Saturated and Subcooled Liquid Small Break Flow Areas for Leak-ECCS Mass Balance Versus RCS Pressure

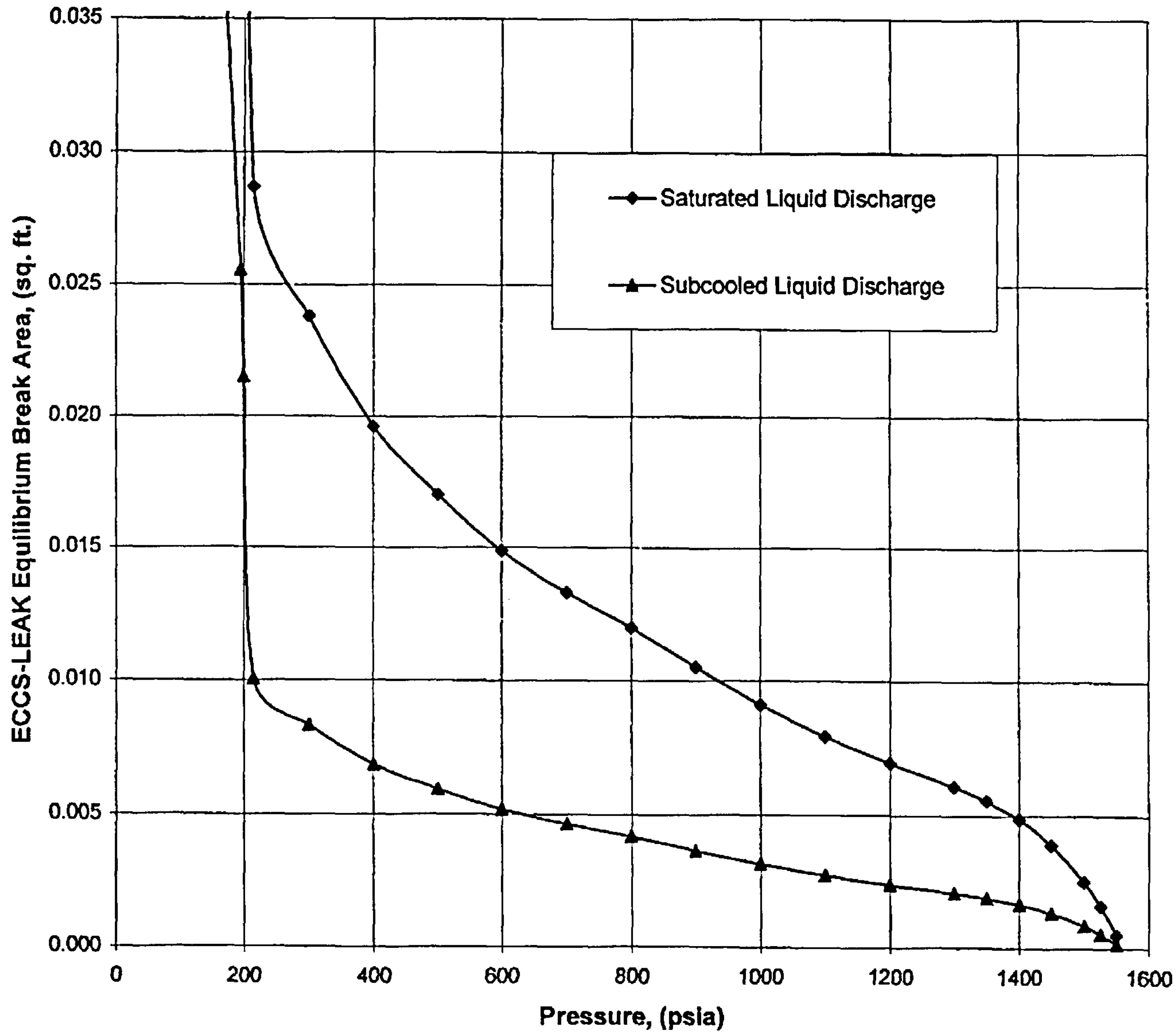
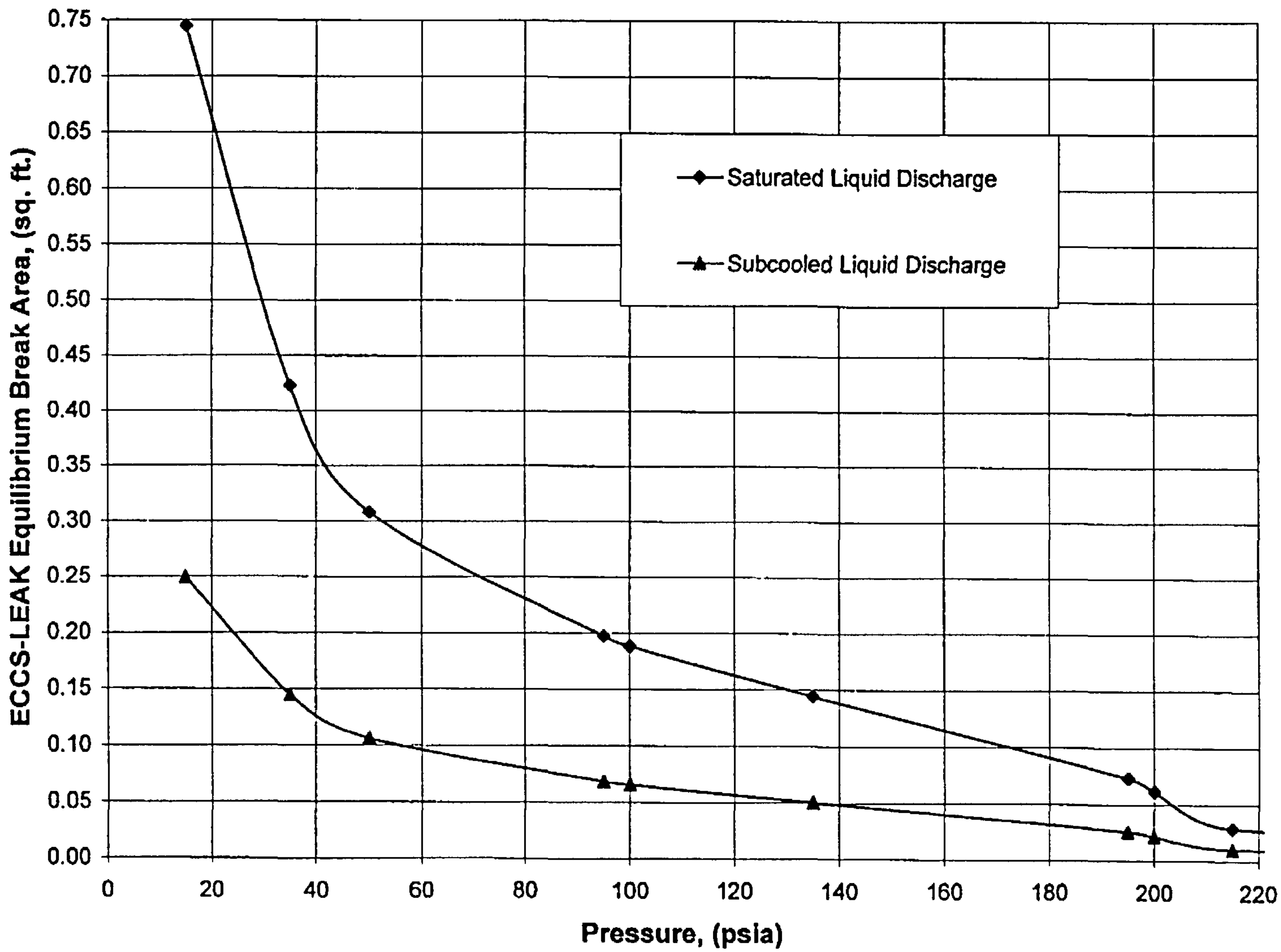


Figure A-3. Saturated and Subcooled Liquid Larger Break Flow Areas for Leak-ECCS Mass Balance Versus RCS Pressure



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
Should FENOC determine that a change to the Mode 3 entry date will affect the exigent circumstances of this application, FENOC management will promptly notify the NRC staff.	Promptly, should the Mode 3 entry date affect the exigency.
Prior to Mode 3 entry, the RCS will be pressurized to approximately 250 psig. The RCS temperature will remain less than 200 °F during the 250 psig inspection (i.e., Mode 5). Visual inspections for RCS leakage will be performed on various RCS components, including the reactor vessel incore monitoring instrumentation (IMI) nozzles.	Prior to Mode 3 operation under the exception proposed by this license amendment application.
If IMI nozzle leakage is discovered, the proposed exception would not be utilized for a Mode 3 entry following corrective action.	Ongoing.
Depending on the postulated break size, assumed single-failure, and plant response, additional operator actions include the following, for smaller break LOCAs, a CFT line break, and long-term boron precipitation control, respectively: 1. For smaller break sizes, depressurization of the RCS to LPI pressure before BWST switchover will be performed. In some cases, this could involve throttling of HPI without maintaining subcooling margin. 2. Open the LPI cross-tie line to provide flow to both CFT lines, or initiate LPI flow through the auxiliary pressurizer spray line. 3. Initiate post-LOCA reactor core boron precipitation control as necessary with LPI flow through the auxiliary	Prior to Mode 3 operation under the exception proposed by this license amendment application.

COMMITMENTS	DUE DATE
pressurizer spray line. These actions are reasonable based on the time expected to be available to diagnose the situation and take the appropriate action. These operator actions will be described in the appropriate plant procedures prior to Mode 3 operation under the proposed license amendment exception.	
Licensed operators will be trained to these new operator actions prior to standing shift in Mode 3 under the exception proposed by this license amendment application.	Prior to standing shift in Mode 3 under the exception proposed by this license amendment application.
During the period under which the proposed exception is effective, the following administrative controls will be implemented as additional compensatory measures: 1. The DBNPS will maximize the availability of plant 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). Required surveillance testing will continue to be performed, however, maintenance activities that adversely affect operability will not be performed. The systems and components include: Low Pressure Injection, Decay Heat Removal, Emergency Diesel Generators, Auxiliary Feedwater, 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. If any of these systems become inoperable for reasons other than performance of surveillance testing, or if the High Pressure Injection system becomes inoperable for reasons other than the capability of maintaining suction from the containment emergency sump (via the LPI system), the plant will initiate a cooldown within two hours to at least Mode 4, at a cooldown rate prudent for the plant conditions.	Prior to Mode 3 operation under the exception proposed by this license amendment application.

COMMITMENTS	DUE DATE
2. The DBNPS will limit activities in the plant's offsite power switchyard and electrical switchgear rooms to those of an essential nature. 3. An additional dedicated licensed operator (above the minimum TS manning requirement) will be on-shift in the control room to assist with the added operator actions which may be necessary to mitigate a Mode 3 LOCA without reliance on the HPI pumps drawing suction from the containment emergency sump (via the LPI pumps). 4. Core reactivity will be controlled in a safe shutdown condition by ensuring that the boron concentration is greater than or equal to the estimated "all rods out" critical boron concentration. Either the control rod groups will be fully inserted with trip breakers open, or a single safety group of control rods will be withdrawn to provide a prompt source of trippable negative reactivity. In addition, potential sources of boron reduction will be strictly controlled to ensure an inadvertent boron reduction does not take place (e.g. demineralized water sources, non-boron saturated purification demineralizers). Boric acid concentration will be sampled at least once per 12 hour shift. Source range nuclear instrumentation count rate will be logged at least once per 4 hours. 5. Operating personnel will be notified by Standing Order to ensure that the above actions are maintained.	