FirstEnergy

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

Guy G. Campbell
Vice President - Nuclear

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

License Number NPF-3

Docket Number 50-346

Serial Number 2633

March 15, 2000

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555-0001

Subject: Request for Exemption from 10 CFR 50, Appendix K, for Boric Acid Precipitation Control Methodology (TAC No. MA7831)

Ladies and Gentlemen:

This letter transmits a request for an exemption from Title 10 of the Code of Federal Regulations (CFR), Section 50, Appendix K, "ECCS Evaluation Models," for the Davis-Besse Nuclear Power Station (DBNPS) boric acid precipitation control (BPC) methodology.

Pursuant to 10 CFR 50.46(a)(1)(ii), the DBNPS Emergency Core Cooling System (ECCS) is modeled in conformance with the required and acceptable features of 10 CFR 50 Appendix K. Appendix K Section I.D.1, "Single Failure Criterion," requires an analysis of possible ECCS equipment failure modes and their effects on ECCS performance during the post-blowdown phase of a postulated loss-of-coolant accident (LOCA). In addition, it requires that the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

The DBNPS is planning to implement a plant modification during the upcoming twelfth refueling outage, which is scheduled to commence in April, 2000. This plant modification will make significant improvements in the post-LOCA BPC methodology by providing two new active means of preventing boric acid precipitation within the reactor vessel core region. However, since there are known single failure vulnerabilities with the methodology, an exemption from 10 CFR 50 Appendix K Section I.D.1 is required in order to credit the methodology per 10 CFR 50.46(a)(1)(ii).

A001

Information supporting this exemption request is contained in Enclosure 1. The DBNPS has discussed the schedule and the technical basis and details of the exemption request with the NRC technical staff in conference calls starting on January 7, 2000. As a result of those discussions, the DBNPS has prepared and included information to address specific issues raised by the NRC staff. As described in Enclosure 1, the DBNPS has concluded that for the reasons specified in the enclosure, special circumstances, as defined in 10 CFR 50.12 exist; that the granting of the requested exemption will not present an undue risk to the health and safety of the public; and that the granting of the requested exemption is consistent with the common defense and security.

It is the DBNPS' goal to complete the plant modification during the upcoming twelfth refueling outage, to closeout the issue of post-LOCA BPC for the DBNPS as a licensing action. The exemption is a necessary part of the completion of this modification. Based on the discussions conducted to date with the NRC technical staff and the additional information now included up-front with the exemption request, the DBNPS requests that the exemption be approved by the NRC by May 1, 2000.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Very truly yours,

Guy G. Campbell

MKL/laj

Enclosures

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

**REQUEST FOR EXEMPTION
FROM
10 CFR 50, APPENDIX K, SECTION I.D.1
SINGLE FAILURE CRITERION**

In accordance with 10 CFR 50.12, "Specific Exemptions," the Davis-Besse Nuclear Power Station (DBNPS) requests an exemption from the requirements specified in 10 CFR 50, Appendix K, "ECCS Evaluation Models," Section I.D.1, "Single Failure Criterion," for the DBNPS boric acid precipitation control (BPC) methodology.

BACKGROUND

Babcock & Wilcox (B&W) designed reactors, such as the DBNPS, require methods to dilute the boric acid which may concentrate in the reactor core following a Reactor Coolant System (RCS) cold leg loss-of-coolant accident (LOCA). Without this BPC methodology, boric acid could concentrate to the point of precipitation, potentially blocking reactor core flow channels and thereby interfering with cooling of the fuel rods. The DBNPS was originally designed with one short-term passive BPC method and two long-term active BPC methods.

The short-term passive method relied on the reactor vessel internal vent valves (RVVVs). The RVVVs provide a flow path between the reactor vessel upper plenum region above the core, and the upper downcomer region, in the event of a postulated cold leg LOCA. This method assumed that the froth level in the reactor vessel upper plenum would spill steam, as well as liquid water with a higher boric acid concentration, through the RVVVs to the cold leg, allowing low pressure injection (LPI) water with a lower boric acid concentration to enter the core, thereby limiting the boric acid concentration in the core and preventing boric acid precipitation in the reactor vessel core region.

The primary long-term active method utilized the decay heat (DH) drop line flowpath, via valves DH-11 and DH-12, from the RCS hot leg (see Attachment 1, Figure 1). This flowpath would provide flow from the RCS hot leg to the suction of a Decay Heat Removal (DHR)/LPI Pump. When used for BPC purposes, flow from the drop line combines with flow from the containment emergency sump (which will be at a lower boric acid concentration) upstream of the DHR/LPI pump, and is then pumped back to the reactor vessel, effectively limiting the boric acid concentration in the core. Since this drop line method is not single failure proof, an alternate long-term active method of dilution, known as RCS hot leg injection, was provided.

The RCS hot leg injection method involved use of a DHR/LPI Pump to inject water from the containment emergency sump into the pressurizer via the auxiliary spray line (see Attachment 1, Figure 2). This lower boric acid concentration water would then flow through the surge line, down the RCS hot leg, to the outlet plenum of the reactor vessel. When the hot leg injection flow exceeds the core boil-off rate, the excess auxiliary spray would cause reverse flow through the core, into the downcomer and out the RCS cold leg break. This reverse flow would maintain the core boric acid concentration below the solubility limit.

In 1991, B&W identified that the analysis supporting the boric acid dilution methods described above had not adequately modeled the reactor vessel internals, rendering the passive BPC method (RVVVs) potentially ineffective. The passive BPC method was needed for a period of time to allow decay heat to fall into a range where available RCS hot leg injection capacity was adequate. Therefore, the RCS hot leg injection method of BPC was determined to be potentially ineffective for LOCAs following plant operation above 23% of rated thermal power. However, further analysis revealed that a previously uncredited flowpath could provide sufficient dilution flow. As part of the reactor vessel design, there are gaps between the reactor vessel outlet nozzles and the core support shield. These gaps provide a flowpath between the outlet annulus and the inlet nozzle/downcomer region of the vessel. From this region, the liquid can reach the RCS cold leg break location. This provides a method of diluting the coolant that remains in the core region. These circumstances were reported to the NRC in DBNPS Licensee Event Report (LER) 91-006, dated December 5, 1991.

The existence of the gaps and the flow that could occur through them were known to exist at the time of plant construction. Topical Report BAW-10091, Supplement 1, "Supplementary and Supporting Documentation for B&W's ECCS Evaluation Model Report with Specific Application to 177-FA Class Plants with Lowered-Loop Arrangement," dated December, 1974, which was submitted to the NRC, discussed the gaps. However, because two active methods of boric acid dilution were believed to be adequate, the capability of the gap flow was not further substantiated. Therefore, flow through the gaps was not originally credited as a post-LOCA boric acid dilution method.

B&W analysis (51-1206351-00), "Long Term Boron Dilution Following Large LOCA Accidents", dated January 16, 1992, evaluated the as-built gaps. Using a conservative gap size and conservative assumptions and initial conditions, it was determined that there is sufficient flow through the gaps to keep the boric acid concentration in the core region well below the boric acid solubility limit. The analysis concluded ". . . the results of the analyses, crediting the gap recirculation flows, show adequate core boric acid concentration dilution without any additional operator initiated systems." However, it is considered prudent to have active measures available so that performance can be monitored.

In 1998, it was identified that for a limited range of small-break LOCA scenarios, initiation of the DHR drop line boric acid dilution flowpath could cause steam binding in the suction piping of both DHR/LPI pumps. These circumstances were reported to the NRC in DBNPS LER 98-008, dated October 1, 1998.

Two active methods of BPC are preferred. As part of the corrective action for LER 98-008, the DBNPS committed to address all issues related to long-term post-LOCA boron dilution, and to complete a related plant modification (described below) by the end of the twelfth refueling outage (12RFO).

PLANNED PLANT MODIFICATION

In order to complete the above-mentioned corrective action identified in LER 98-008, a plant modification is planned for implementation during 12RFO. The 12RFO is scheduled to commence in April, 2000. This modification will establish new primary and backup active BPC methods.

Primary Method Description: Improved Auxiliary Spray Method

An improved auxiliary pressurizer spray method will serve as the new primary active BPC method (see Attachment 1, Figure 3). Under the current backup BPC method, a DHR/LPI Pump is utilized to supply water to the pressurizer via the auxiliary spray line. The new method will utilize High Pressure Injection (HPI) Pump 2 to supply water to the auxiliary spray line via a newly installed tie-line. Use of an HPI pump rather than a DHR/LPI pump for the auxiliary spray BPC method will provide a much greater flow rate to the pressurizer. This flow rate will be effective for any size LOCA initiated at 102% of the current licensed power level of 2772 MWt. The capability of the improved auxiliary pressurizer spray method has been evaluated taking into consideration the one hour (approximate) time required to refill the pressurizer.

The new 2-1/2 inch (nominal) tie-line, including two 2-1/2 inch (nominal) manual gate valves, HP-209 and HP-210, will connect into the auxiliary spray flowpath between existing valve DH-178 and flow element FE4999. In addition, as shown in Figure 3, two 1-1/2 inch (nominal) manual gate valves, DH-200 and DH-201, will be installed in the DHR system between valve DH-178 and the connection to the new tie-line. During power operation, the systems will be lined-up in the BPC mode, with valves DH-200 and DH-201 closed and valves HP-209 and HP-210 opened. During normal shutdown cooling operation, valves DH-200 and DH-201 will be opened and valves HP-209 and HP-210 closed, to allow for normal cooldown of the pressurizer.

Use of the new flowpath will require that the HPI injection valves be closed, i.e., the full available HPI Pump 2 capacity (less recirculation) will be available for injection into the pressurizer for BPC. In the BPC mode, HPI Pump 2 will be capable of discharging water at approximately 1700 psig (approximately the shutoff head of the pump), taking "piggyback" suction from the emergency sump via DHR/LPI Pump 2. The flow rate will be sufficient (approximately 300 gpm) to exceed the core boiloff rate prior to the boric acid concentration reaching the solubility limit. The excess spray flow initiates a reverse core flow that will reduce the core boric acid concentration and preclude potential precipitation concerns by transporting the fluid with high boric acid concentrations backward through the downcomer and out the break.

As part of the plant modification, due to increased flow and pressure requirements, a portion of the existing auxiliary spray line, including the portion containing auxiliary spray flow instrumentation FE4999, will be re-sized and upgraded. The new FE4999 flow indication loop will be qualified for the post-LOCA environment, providing suitable indication to allow control room operators to monitor the success of the BPC flowpath.

Framatome Technologies, Inc. (FTI), the B&W successor company, has calculated that there is insufficient boric acid available in the entire RCS and Emergency Core Cooling System (ECCS) to reach the solubility limit if the RCS is above 322 °F. If the RCS is below 322 °F, but subcooled, no boiling is occurring in the core region and therefore boric acid concentration is not of concern. Therefore, BPC is not required when the core exit is adequately subcooled. Consequently, prior to initiation of this primary BPC method, the operator will confirm the need for BPC based on RCS conditions. A DBNPS calculation has determined that the average of 13 incore thermocouple instruments, as read in the Control Room, or on the plant computer, will be of sufficient accuracy to determine the RCS is still above 322 °F, when an averaged reading of 333 °F is obtained (this includes instrument errors). The incore thermocouples are assured to be available following an accident. It can be determined that the RCS is at saturation conditions by trending RCS pressure and temperature, and verifying that the readings are close to the saturation curve and that the trend parallels the saturation curve. This information is available from the Safety Parameter Display System (SPDS). Since this indication of saturated conditions is not dependent on a specific setpoint, there is no need to include instrument error considerations. All the instrumentation associated with this decision process is environmentally qualified, so that it will be available following a LOCA that requires BPC.

Because this primary BPC method is only connected to one train of HPI, it is subject to failure if any single active component in the flowpath fails. This can be due to either a mechanical or an electrical failure. In this event, the backup method of BPC (described below) would be implemented, if not affected by the primary BPC method failure.

Backup Method Description: Improved DHR Drop Line Method

The new backup active BPC method (see Attachment 1, Figure 4) will utilize a flowpath similar to the current DHR drop line primary method. The new backup method is also similar to the second method given in Section 10.6.2 of B&W Topical Report BAW-10105, "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS", Revision 1. One of the two operating DHR/LPI pumps will take suction from the DHR drop line and will discharge a low (throttled) flow rate into the reactor vessel via the core flood nozzles. The second pump will be unthrottled and will continue to take suction from the emergency sump. The flow through the drop line will allow forward flow through the reactor vessel, so that any amount of flow of relatively low concentration water from the train aligned to the sump will enter and dilute the boric acid in the core. FTI has performed calculations to define conditions and flow rates that will not result in vapor entrainment for the pump taking suction from the drop line. Specific guidance will be provided to control room operators to ensure that no vapor entrainment will damage the DHR/LPI pump. The existing system throttle valves will be used to provide flow control.

The new backup BPC method differs from the existing primary BPC method in several ways. The existing method aligns the suction of a DHR/LPI pump to both the DHR drop line and the emergency sump, whereas the new method will have one DHR/LPI pump take suction exclusively on the drop line (i.e., in DHR mode), while the other DHR/LPI pump takes suction on the emergency sump (i.e., in LPI mode). Therefore, with the existing method, the dilution actually occurs upstream of the pump, whereas with the new method, the dilution will occur in the reactor vessel downcomer region.

With the existing BPC primary method, DHR/LPI pump suction valves DH-1517 and DH-1518 remain closed, and existing lines to bypass these valves are utilized. Maintaining DH-1517 and DH-1518 closed could result in a large differential pressure across the bypass lines, with resultant flashing and steam admission to the DHR/LPI pump suctions. However, with the new BPC method, DH-1517 or DH-1518 will be fully open and throttling will occur downstream of the DHR cooler, preventing flashing in any part of the system. Therefore, the plant modification includes changes to valves DH-1517 and DH-1518 to ensure they are capable of opening post- LOCA for BPC purposes. Since these valves could potentially be subject to pressure locking due to ambient temperature changes attributable to the LOCA, a pressure equalization path will be added from between the valve disks to the RCS side of each valve, as part of the planned plant modification. The valves and their motor operators will meet the requirements of Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," dated August 17, 1995. Also, as part of the plant modification, the bypasses around valves DH-1517 and DH-1518 will be isolated via closure of valves DH-10 and DH-26, and the existing flow indication in each bypass line will no longer be available.

As a part of the supporting analysis of the backup BPC methodology, FTI determined that the boric acid concentration in the DHR drop line coming from the core could exceed the solubility limits for the temperatures found in the DHR cooler. This concern only exists for the backup BPC method, since the flowpath for the primary method comes from the relatively dilute emergency sump. FTI performed additional analyses to demonstrate that boric acid precipitation in the DHR cooler will not occur. The details of this additional analyses are provided in Attachment 3.

The new backup BPC method would only be utilized if the new primary method was unavailable and if both DHR/LPI pumps were functioning. Motor-operated valves DH-11 and DH-12 are located in a water-tight pit in the lower elevation of containment (which will be flooded post-LOCA). Initiation of the backup method soon after switchover from the borated water storage tank (BWST) to the ECCS emergency sump preserves the viability of the backup method should the primary method fail and the water-tight pit leak excessively. Therefore, valves DH-11 and DH-12 will be opened after sump switchover, provided the RCS is within the design pressure and temperature range for the DHR drop line piping and components. FTI has shown that RCS cold leg pump discharge break sizes of 0.09 ft^2 and larger will cooldown (without operator assistance) below the DHR drop line design range by the time of sump switchover. Smaller breaks may not evolve to these conditions at the time of sump switchover, however, they will allow the flow from both DHR/LPI pumps to refill the reactor vessel such that passive core boric acid dilution is obtained through RVVV liquid spillover. The reactor vessel refill will occur prior to the RCS reaching the DHR drop line initiation range or the core solubility limit. Once RVVV liquid dilution has been established, it will halt the core boron concentration increase, and liquid through-put will begin to dilute the core concentration. The liquid through-put will prevent precipitation in the core and reduce the core concentration below the decay heat cooler solubility limit until the DHR drop line design conditions are reached and the backup method can be established.

<u>Common Mode Failure Vulnerabilities</u>

To support the planned plant modification, a failure modes and effects analysis (FMEA) was performed to identify any common-mode failure vulnerabilities between the two new active BPC methods. The analysis consisted of six sections:

- Single Failure Analysis – Primary Long-Term BPC Flowpath
- Single Failure Analysis – Alternate Long-Term BPC Flowpath
- Electrical Failure Analysis - 480 Essential Motor Control Centers (MCCs) - Train 1
- Electrical Failure Analysis - AC Distribution Buses and Emergency Onsite Power - Train 1
- Electrical Failure Analysis - 480 Essential MCCs - Train 2
- Electrical Failure Analysis - AC Distribution Buses and Emergency Onsite Power - Train 2

The analysis considered potential failure modes for components that perform an active mechanical function or an active or passive electrical function. The failure modes for each active component and the effect a failure has on the ability of the primary and/or alternate (backup) long-term BPC flowpaths to perform the intended function were identified. The following common-mode failure vulnerabilities were identified through the FMEA:

1. The primary BPC method utilizes a new tie-line that will be available for use only with HPI Train 2. In addition, both DHR/LPI trains must be available in order to use the backup BPC method, since one train must continue to perform the LPI function

while the other train performs the BPC function. Therefore, since both the primary and alternate (backup) long-term BPC flowpaths rely on Train 2 components, failure modes that affect both HPI Train 2 and DHR/LPI Train 2 result in the loss of both the primary and alternate long-term BPC flowpaths. Potential failures of concern are:

- Emergency Diesel Generator (EDG) 2 failure following a loss of offsite power
- 4160 V essential bus D1 lockout
- Loss of DC control power to 4160 V essential bus D1
- Loss of Service Water (SW) Train 2 (including ventilation)
- Loss of Component Cooling Water (CCW) Train 2 (including ventilation)
- Loss of ECCS Room Coolers 1 and 2 (serving ECCS Room containing Train 2 DHR/LPI and HPI pumps)

It is of note that a failure of EDG 2 could be mitigated by use of the non-class 1E Station Blackout Diesel Generator (SBODG). The SBODG is described in the DBNPS Updated Safety Analysis Report (USAR) Section 8.3.1.1.4.2, "Alternate AC Source - Station Blackout Diesel Generator."

It is also important to note that, as described in USAR Section 9.2.1, "Service Water System," three Service Water (SW) pumps are provided, however only two are required for operation. SW Pumps 1 and 2 are powered from 4160 V essential buses C1 and D1, respectively. SW Pump 3, the "spare" pump, can be powered from either C1 or D1. Similarly, as described in USAR Section 9.2.2, "Component Cooling Water," three CCW pumps and heat exchangers are provided, however only two are required for operation. CCW Pumps 1 and 2 are powered from 4160 V essential buses C1 and D1, respectively, and CCW Pump 3, the "spare" pump, can be powered from either C1 or D1. Therefore, a pump failure in SW or CCW Train 2 could be mitigated by use of the spare SW or CCW pump, if available.

Finally, the primary BPC flowpath includes two motor-operated valves in series, DH-2735 and DH-2736, which are powered from Train 1 and Train 2 essential power, respectively. Similarly, the backup BPC flowpath includes two motor-operated valves in series, DH-11 and DH-12, which are powered from Train 2 and Train 1 essential power, respectively. Valves DH-2735 and DH-12 are powered from 480 V essential MCC E11B. Valves DH-2736 and DH-11 are powered from 480 V essential MCC F11A. In case of an electrical failure, the capability is provided to interconnect MCCs E11B and F11A, so that all the valves can be powered from the unaffected power supply. Both the existing and the proposed system require cross tie capability. This capability is described in USAR Section 8.3.1.1.11, "Cross Ties Between MCC E11B and F11A." The actions to accomplish the interconnection are presently in plant procedures.

2. A single failure of valve DH-9A, valve DH-2734, or DHR/LPI Pump 2 would result in the temporary loss of both the primary and alternate flowpath. However, the DHR Train 1 cross tie alignment could be performed by opening 8 inch (nominal) motor-operated valve DH-831, which receives Train 1 essential power. (Note that motor-operated valve DH-830, which receives Train 2 essential power, is not required to open. With DH-831 open, flow through the cross-tie bypasses valve DH-830 via two 8 inch (nominal) check valves.) In this alignment, DHR/LPI Pump 1 would supply suction head to HPI Pump 2. This decay heat cross tie alignment is discussed in USAR Section 6.3.2.11, "Reliability Considerations," and USAR Table 6.3-6, "Single Failure Analysis - Emergency Core Cooling System," and the actions to accomplish this alignment are presently in plant procedures.

REASON FOR EXEMPTION REQUEST

Pursuant to 10 CFR 50.46(a)(1)(ii), the DBNPS ECCS is modeled in conformance with the required and acceptable features of 10 CFR 50 Appendix K. Appendix K Section I.D.1, "Single Failure Criterion," requires an analysis of possible ECCS equipment failure modes and their effects on ECCS performance during the post-blowdown phase of a postulated LOCA. In addition, it requires that the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

As previously described, the DBNPS is planning to implement a plant modification during the upcoming twelfth refueling outage, which is scheduled to commence in April, 2000. This plant modification will make significant improvements in the post-LOCA BPC methodology by providing two active means of preventing boric acid precipitation. However, since there are known single failure vulnerabilities that would disable both new active modes of BPC, an exemption from 10 CFR 50 Appendix K Section I.D.1 is required for crediting this plant modification as meeting 10 CFR 50.46(a)(1)(ii).

JUSTIFICATION FOR EXEMPTION

10 CFR 50.12(a) states the Commission may grant an exemption from requirements contained in 10 CFR Part 50 provided that the exemption (1) is authorized by law, (2) will not present an undue risk to public health and safety, (3) is consistent with the common defense and security, and (4) special circumstances are present.

The requested exemption from the single failure requirement of 10 CFR 50 Appendix K, Section I.D.1, satisfies these requirements, as described below:

1. The requested exemption is authorized by law

The NRC authority to grant exemptions from the requirements of Title 10 of the Code of Federal Regulations, Part 50 is codified in 10 CFR 50.12. Since the exemption request does not present an undue risk to public health and safety, and will not endanger the common defense and security, as discussed below, the NRC is authorized to issue the exemption.

2. The requested exemption will not present an undue risk to public health and safety

A risk evaluation was performed to determine the Core Damage Frequency (CDF) due to an accident occurring, in combination with a failure that renders both active BPC methods inoperable. The results of the risk evaluation show that the CDF contribution due to such an occurrence would be very low, of the order of 1.1×10^{-7} per reactor year. The BPC failure mode was not previously modeled in the DBNPS' plant risk assessment because BPC was viewed as a long-term action, and as such, its failure was considered to be beyond the scope of the analysis. Therefore, it is not possible to quantify the decrease in overall plant CDF due to the planned BPC plant modification. However, when compared to the current overall plant CDF of approximately 1.63×10^{-5} per reactor year, it is clear from the risk evaluation that BPC failure is not a significant risk contributor. The risk evaluation is conservative in that it does not consider any long-term corrective actions that could restore the BPC function in the event of its failure.

Since the planned BPC plant modification adds a tie-line from only HPI Train 2 to the auxiliary spray line for the primary BPC method, an evaluation was performed to compare the previous case (described above) with the decrease in CDF that would occur if the scope of the planned plant modification was expanded to include a similar tie-line for HPI Train 1. Such an increase in scope could address the common-mode failure mechanism concerns that exist between HPI Train 2 and DHR/LPI Train 2. However, the evaluation determined that the CDF contribution due to an accident occurring, in combination with a failure that renders all active BPC methods inoperable (including the assumed additional primary method) was less than an order of magnitude lower compared to the previous case, specifically 1.3×10^{-8} per reactor year. The effect of such an increase in scope of the planned plant modification is not considered to be significant under the guidance of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July, 1998, therefore such an increase in scope is not considered to be warranted.

The frequency of a Large Early Release (LERF) to the environment due to a containment failure was also examined. As in the case of the CDF evaluation, this event had not been previously considered, so comparison of the change can not be made. However, LERF was found to be 1.1×10^{-11} per reactor year for the planned plant modification. When compared to the overall plant LERF of 7.3×10^{-8} per reactor year, it is clear that the BPC failure is not a significant risk contributor. This result is consistent with the types of scenarios that can lead to Large Early Releases, and the times required for boric acid precipitation to occur. Therefore, while enhancement of the boric acid control measures does reduce the LERF, it would not be expected to represent a significant portion of the overall LERF. The LERF that would be obtained by adding a second HPI train to the plant modification scope would be improved by 8.4×10^{-12} per reactor year. This is deemed to be an insignificant improvement.

Additional detail regarding the CDF and LERF evaluation is provided in Attachment 2.

It is also noted that the calculations that validate the effectiveness of the two new active BPC methods contain numerous conservative assumptions. For example, Appendix K of 10 CFR 50 requires that a decay heat of 1.2 times the decay heat for an infinitely irradiated core be used in predicting the decay heat of the core. The higher decay heat levels provide more boric acid concentration earlier in the time following the accident. Calculations have shown that using a decay heat level of 1.0 significantly improves the effectiveness of active methods. This in turn increases the time available to restore any equipment failures to the point where either active method will be effective, even if initiated later than required by procedures. In addition, except as noted in the backup method discussion regarding DHR cooler considerations, the calculations take no credit for the passive dilution methods (such as the gaps between the reactor vessel outlet nozzles and the core support shield) that are inherent in the DBNPS design and construction. These conservative factors, in addition to the conservative factors imposed by 10CFR50, Appendix K, provide additional time before BPC would actually be required to be initiated.

In summary, the extremely low probability of experiencing the failures that render both active methods inoperable concurrent with the accident represent an insignificant part of the plant's overall core damage frequency. Ways of mitigating several of the common-mode failure mechanisms have been identified. The conservatism included in the design calculations, as well as the presence of passive boric acid precipitation control mechanisms, all provide assurance that boric acid precipitation will not occur in the post-LOCA environment. It is also important to note that the planned plant modification will result in a substantial improvement in post-LOCA BPC capability. Based on the above, this exemption does not pose an undue risk to the health and safety of the public.

3. <u>The requested exemption is consistent with the common defense and security</u>

To ensure that the common defense and security are not endangered, the exemption request must demonstrate that the loss or diversion of Special Nuclear Material (SNM) is precluded. The DBNPS has systems and processes in place that provide protection for the public from diversion of SNM that is licensed to be possessed on site. These systems and processes are those embodied in the "Davis-Besse Nuclear Power Station Physical Security Plan," the "Davis-Besse Nuclear Power Station Guard Training and Qualification Plan," and the "Davis-Besse Nuclear Power Station Safeguards Contingency Plan."

The request for exemption from the single failure criterion of 10 CFR 50 Appendix K, Section I.D.1, does not affect the systems and processes discussed above. Therefore, this exemption does not affect the common defense or security.

4. Special circumstances are present

10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemption meets the special circumstances of 10 CFR 50(a)(2)(ii), in that the application of these regulations in this circumstance is not necessary to achieve the underlying purpose of the regulations.

The underlying purpose of Appendix K, Section I.D.1, as it relates to 10 CFR 50.46(b)(5), is to assure long-term cooling performance of the ECCS in the event of the "most damaging single failure of ECCS equipment." As discussed above, the DBNPS will have two active BPC methods in place following 12RFO. While the two methods are subject to common failure mechanisms, long-term core cooling can still be assured by effecting repairs or taking alternative actions to mitigate the failure in a timely fashion. Procedural controls will be in place to alert operators to the need for establishing BPC, and to ensure that they respond promptly to restore the BPC function as quickly as possible if a failure occurs that precludes using both methods. Conservative methods of analysis and the presence of passive flowpaths ensure that the active method restoration is highly credible in the time allotted. Although the single failure criterion is not satisfied, measures will be promptly taken to restore the active methods, thereby assuring long-term core cooling. Therefore, the DBNPS concludes that the underlying purpose of Appendix K, Section I.D.1, as it relates to 10 CFR50.46(b)(5), will be achieved.

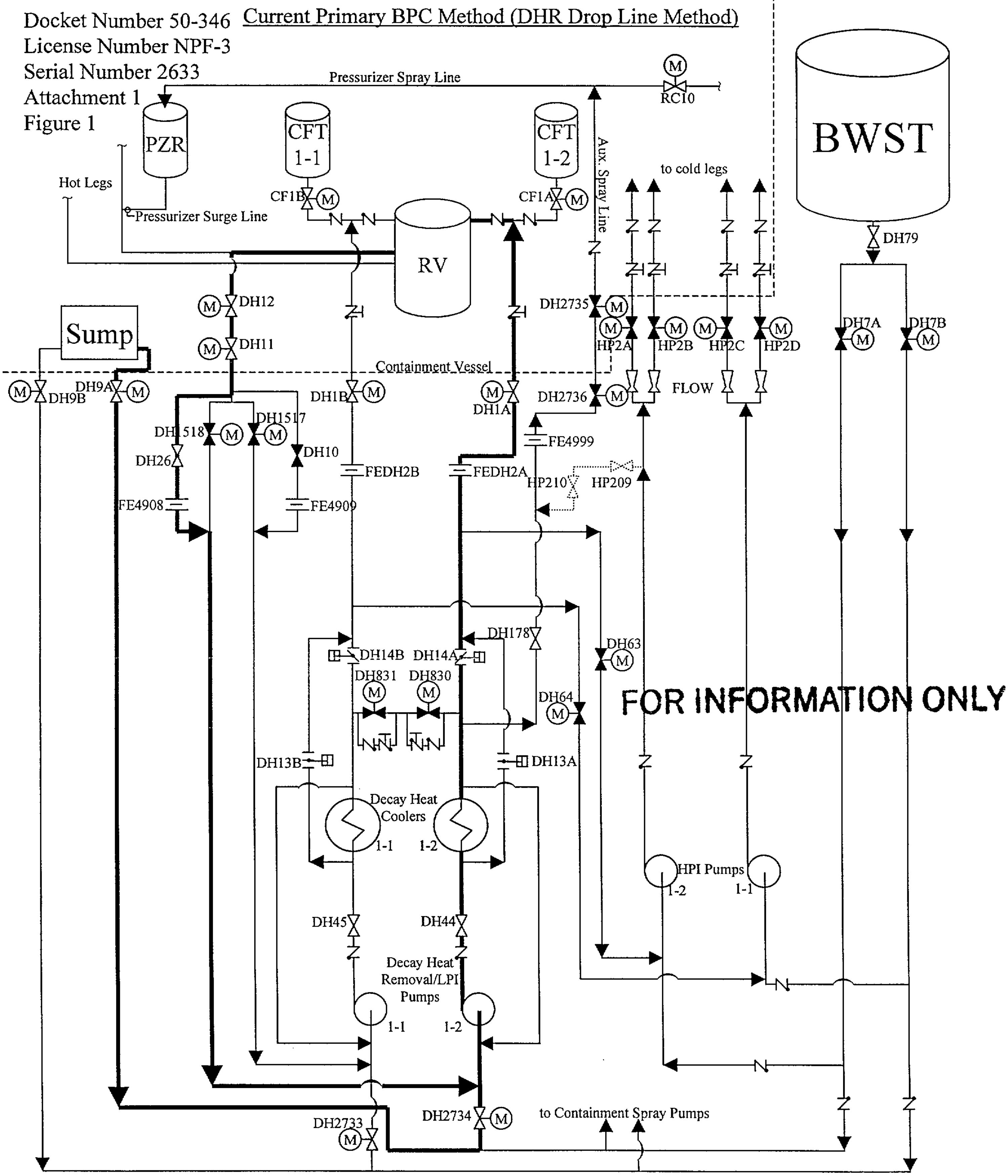
ATTACHMENTS

Attachment 1 - Four Figures:
- Figure 1, "Current Primary BPC Method (DHR Drop Line Method)"
- Figure 2, "Current Backup BPC Method (Auxiliary Spray Method)"
- Figure 3, "Planned Primary BPC Method (Improved Auxiliary Spray Method)"
- Figure 4, "Planned Backup BPC Method (Improved DHR Drop Line Method)"

Attachment 2 - Summary of Boric Acid Precipitation Control Calculation C-NSA-099.16-26

Attachment 3 - Summary of Decay Heat Removal Cooler Evaluation

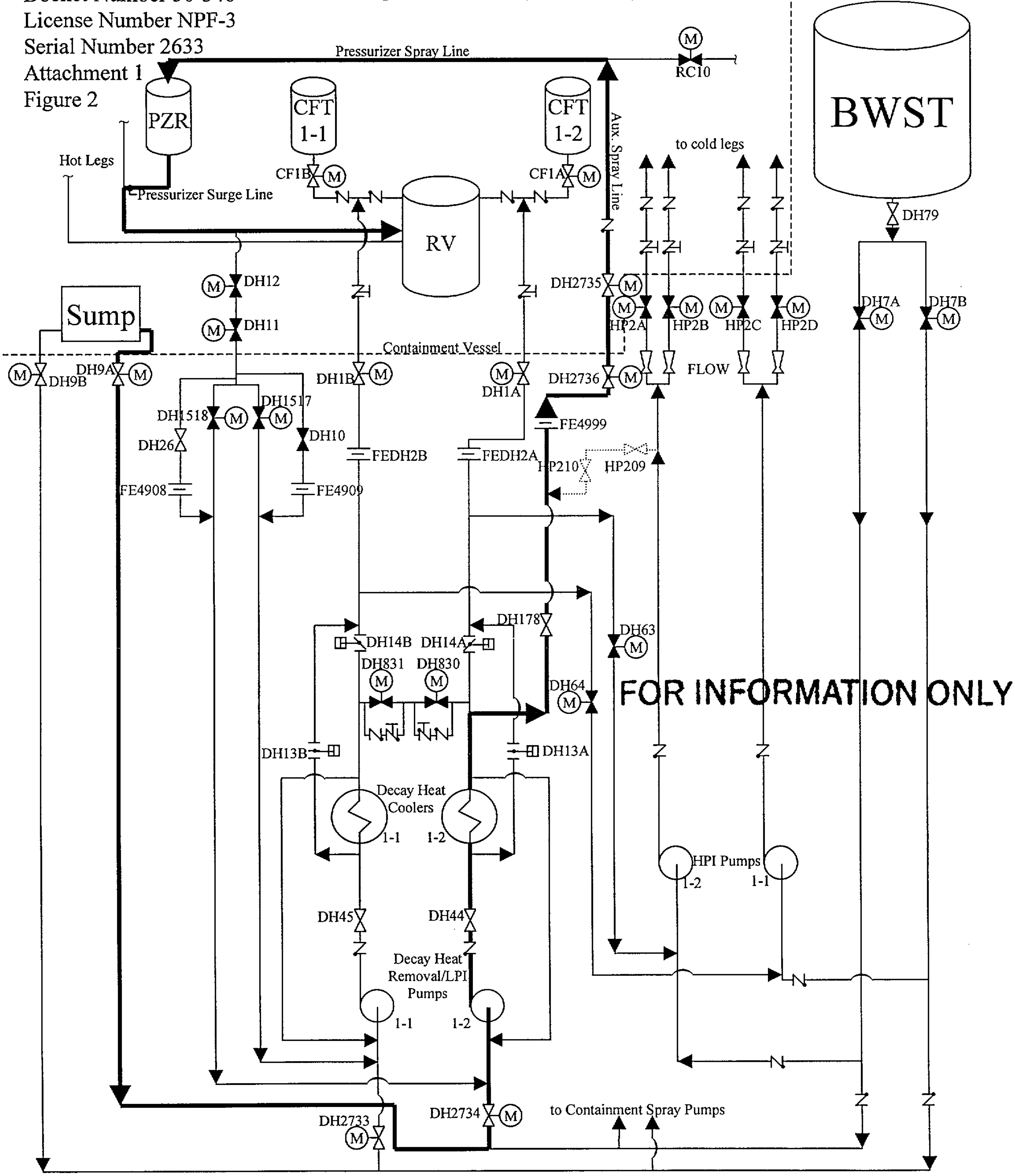
Docket Number 50-346
License Number NPF-3
Serial Number 2633
Attachment 1
Figure 1



Notes
1. Valves shown in long-term post-LOCA line-up, with Train 1 performing LPI function and Train 2 performing BPC/LPI function.
2. Cross-tie containing valves HP 209 and HP 210 is shown per planned plant modification.
3. This is a simplified schematic. Not all flowpaths/components are shown.

Docket Number 50-346
License Number NPF-3
Serial Number 2633
Attachment 1
Figure 2

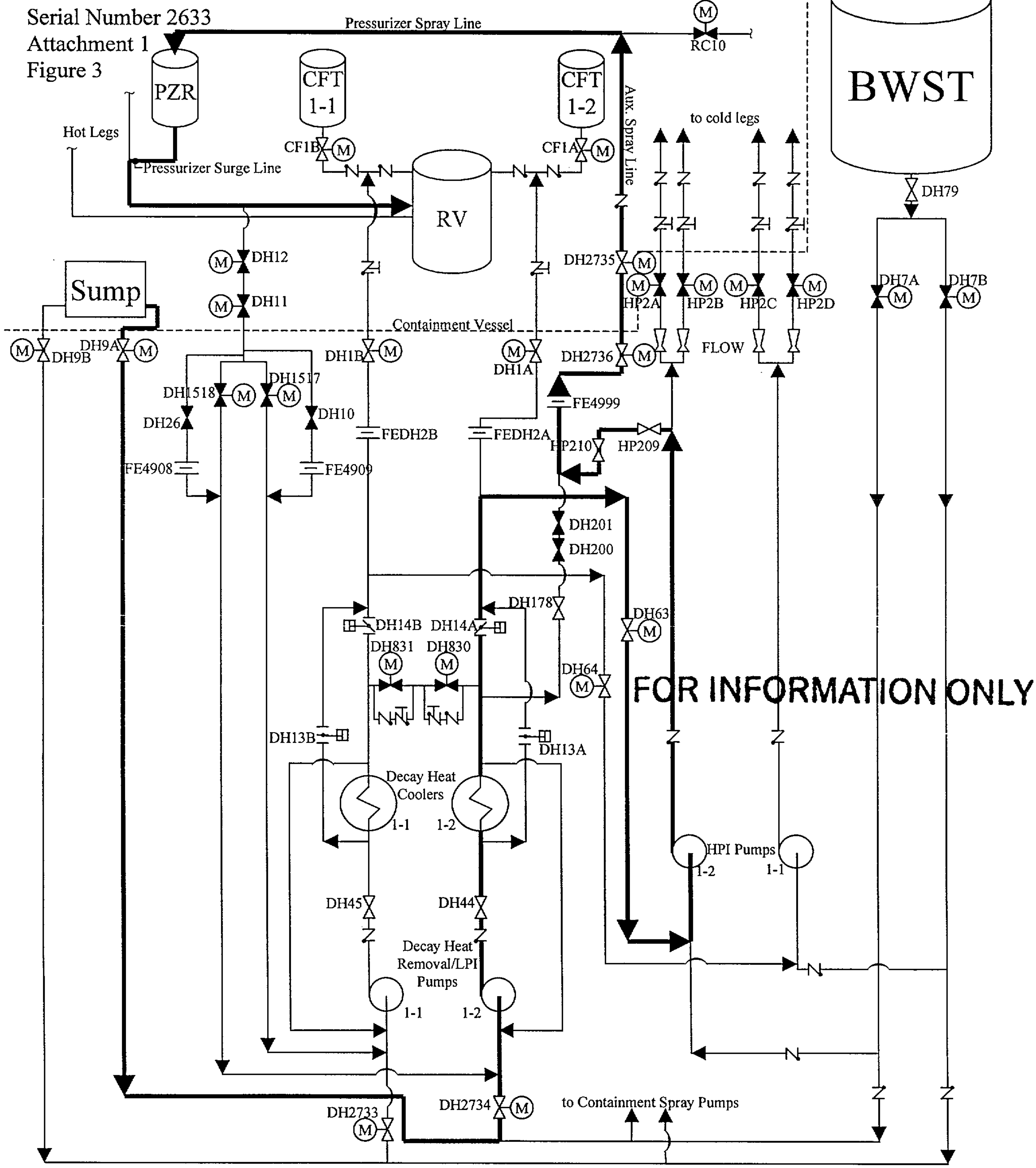
Current Backup BPC Method (Auxiliary Spray Method)



Notes
1. Valves shown in long-term post-LOCA line-up, with Train 1 performing LPI function and Train 2 performing BPC/LPI function.
2. Cross-tie containing valves HP 209 and HP 210 is shown per planned plant modification.
3. This is a simplified schematic. Not all flowpaths/components are shown.

Docket Number 50-346
License Number NPF-3
Serial Number 2633
Attachment 1
Figure 3

Planned Primary BPC Method (Improved Auxiliary Spray Method)

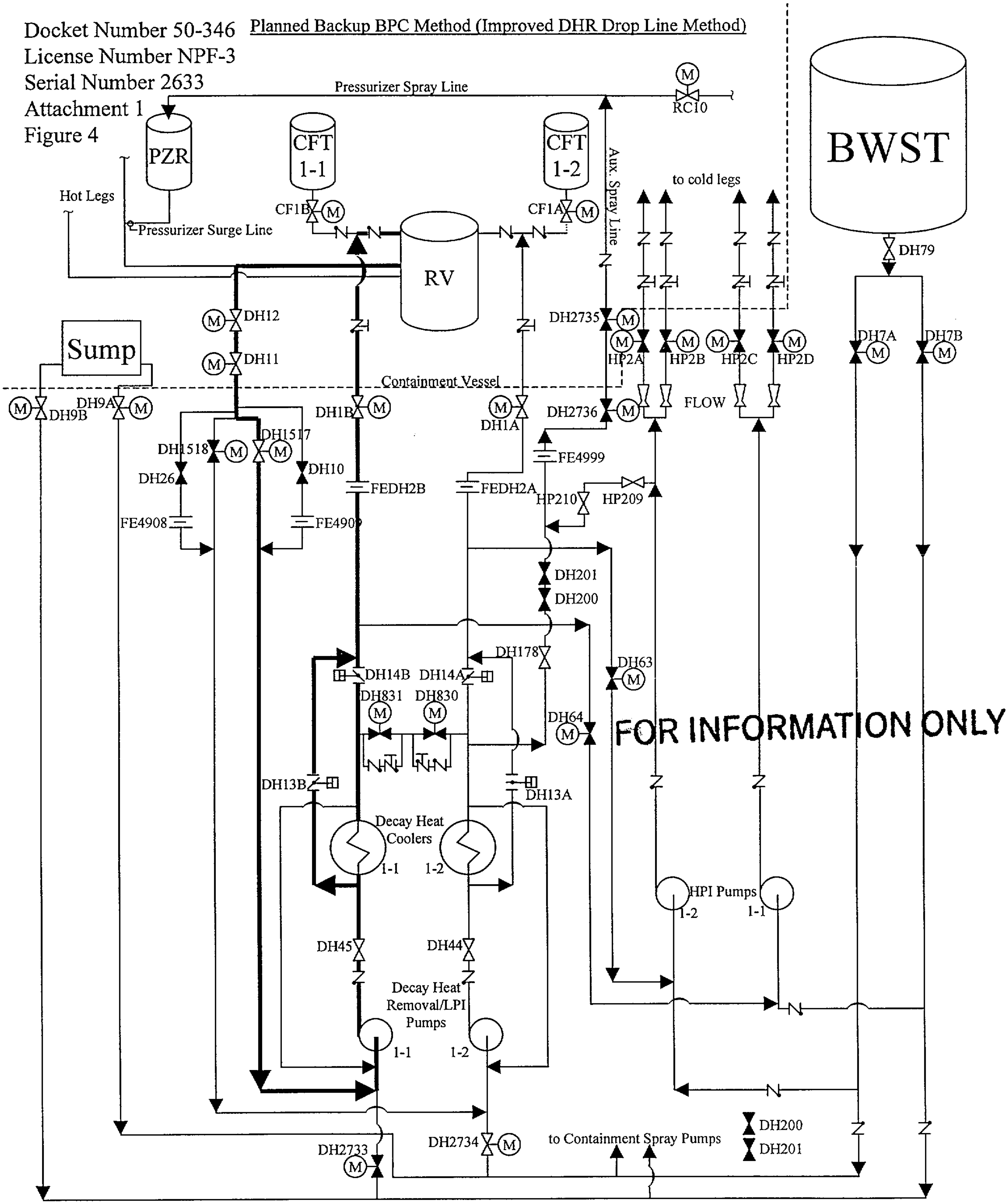


FOR INFORMATION ONLY

Notes
1. Valves shown in long-term post-LOCA lineup, with Train 1 performing LPI function and Train 2 performing BPC/LPI function.
2. Cross-tie containing valves HP209 and HP 210 is shown per planned plant modification.
3. This is a simplified schematic. Not all flowpaths/components are shown.

Docket Number 50-346
License Number NPF-3
Serial Number 2633
Attachment 1
Figure 4

Planned Backup BPC Method (Improved DHR Drop Line Method)



FOR INFORMATION ONLY

Notes
1. Valves shown in long-term post-LOCA line-up, with Train 1 performing BPC function and Train 2 performing LPI function.
2. Cross-tie containing valves HP209 and HP210 is shown per planned plant modification.
3. This is a simplified schematic. Not all flowpaths/components are shown.

Summary of Boric Acid Precipitation Control Calculation C-NSA-099.16-26

Background

The evaluation of risk due to the failure of both active boric acid precipitation control (BPC) methods is documented in Davis-Besse Nuclear Power Station (DBNPS) Calculation C-NSA-099.16-26. This calculation provides a bounding estimate of the risk of core damage and containment release due to the failure of BPC methods after the planned plant modification. Additionally, this calculation evaluated the reduction in the risk that could be gained by the extension of the proposed modification to include a line from High Pressure Injection (HPI) Pump 1-1 to the auxiliary spray line.

Assumptions

The following assumptions were made and documented in the assumptions section of Calculation C-NSA-099.16-26.

1. It was assumed that if the active methods of BPC fail the result will be core damage due to inadequate core cooling. No credit was taken for passive BPC methods mitigating precipitation in the core if the active methods fail after or during attempts to initiate them.

2. It was assumed that active BPC is needed five hours after the event initiation. The primary dilution method must be initiated one hour prior to the time that dilution is needed, to allow time for the pressurizer to fill. Therefore, flow through the primary boric acid dilution flow path was assumed to be required within 4 hours and flow through the alternate flow path was assumed to be required within 5 hours. These times are minimums based on the boric acid precipitation analysis performed by Framatome Technologies Inc. (FTI), which also include significant conservatisms. Therefore, this assumption represents a significant source of conservatism in the human reliability analysis performed for this calculation. A more precise method would have been to use a range of initiating event frequencies with corresponding response times. However, this approach would have added more complexity to the analysis and was judged to be not required for a bounding type calculation.

3. It was assumed that a method of active BPC is required for all large or medium loss-of-coolant accidents (LOCAs). Not all of the medium LOCAs, as defined in the Probabilistic Safety Assessment (PSA), would be in the break size range that requires boric acid dilution flow. Therefore, including all medium LOCA's in the initiating event frequency contributed to a conservative result.

4. Boric acid dilution is only required for breaks lower than the 573 foot elevation in the cold leg reactor coolant pump discharge piping. Therefore, it is assumed that 25% of the large and

medium LOCA frequency involves breaks in locations that would require a boric acid dilution flow path. The 25% fraction was assumed because it should provide a conservative estimate, given the limited portion of the reactor coolant system susceptible to breaks requiring boric acid dilution flow.

Assumptions 3 and 4 both result in a conservative estimate of the initiating event frequency for this analysis. A less conservative, but valid approach would have been to estimate the initiating event frequency using the failure rates published in the *Pipe Failure Study Update*. This reference provides a failure rate of 2.87E-11 per section hour for pressurized water reactor (PWR) reactor coolant system (RCS) piping. The applicable number of pipe sections is four for this analysis and the conditional probability of a large or medium break is .75. Therefore, the initiating event frequency using this approach would be 7.54E-7 per year. This result is significantly less than the initiating event frequency of 1.1E-5 used in the calculation based on assumptions 3 and 4. If an initiating event frequency of 7.54E-7 had been used in Calculation C-NSA-099.16-26, the results would have been:

Table 1. Summary of Results with Modified Initiating Event Frequency

PSA Result	Case 1 Post Modification Configuration		Case 2 Post Modification with Additional Line from HPI Pump 1-1	
	IE Frequency 1.1E-5 / year	IE Frequency 7.5E-7 / year	IE Frequency 1.1E-5 / year	IE Frequency 7.5E-7 / year
CDF	1.1E-7	7.5E-9	1.3E-8	8.9E-10
LERF	1.1E-11	7.5E-13	2.6E-12	1.8E-13

Evaluation of Post Modification Core Damage Frequency (CDF)

Failure of long term boric acid dilution is not modeled in the present PSA revision so a new sequence was developed for this evaluation. This new sequence represents a large or medium LOCA with success of both low pressure injection (LPI) and low pressure recirculation but failure to align one of the two active long term boric acid dilution methods.

A boric acid dilution flowpath is required for LOCAs that are large enough to cause the depressurization of the reactor coolant system. Therefore, this frequency is encompassed by the large and medium LOCA frequencies in the PSA. Initiating event frequencies for generic events, including LOCAs, have recently been published in NUREG/CR-5750. This reference gives a large LOCA frequency of 5E-6 and medium LOCA frequency of 4E-5. However, boric acid dilution is only required for LOCAs in the cold leg piping. Therefore, the LOCAs of concern for boric acid dilution represent only a fraction of the total LOCA frequency. To conservatively account for the relatively small fraction of the total reactor coolant system piping in the cold legs the LOCA frequencies was adjusted by a factor of .25. The total initiating event frequency is then calculated as follows:

Table 2. Calculation C-NSA-099.16-26 Initiating Event Frequencies

Event	Event Frequency Per NUREG/CR-5750	Modified Event Frequency
Large LOCA	5E-6	1.25E-6
Medium LOCA	4E-5	1E-5
Total	4.5E-5	1.1E-5

A fault tree was constructed for this analysis with a top event representing core damage due to post-LOCA boric acid precipitation. As much as possible this fault tree used supporting logic previously developed for the PSA.

Six human actions were developed for the revised fault tree using the methods described in the DBNPS human reliability notebooks. The events and their calculated values are as follows:

Table 3. Human Event Probabilities

Event	Description	Probability
LHALTBPE	Operators fail to initiate long term boric acid dilution through the primary flow path.	9.9E-4
LHALTBAE	Operators fail to initiate long term boric acid dilution through the alternate flow path.	7.9E-4
EHAE11BE	Operators fail to cross connect E11B and F11A after a loss of power to E11B.	3.8E-3
EHAF11AE	Operators fail to cross connect E11B and F11A after a loss of power to F11A.	3.8E-3
LHAXCONE	Operators fail to cross connect LPI and balance flow if an LPI pump fails.	9.9E-4
ZHAC2H4E	Failure to initiate long term boric acid dilution via the primary path (LHALBDPE) and failure to initiate long term boric acid dilution via the alternate path (LHALBDAE)	9.9E-4

The new PSA sequence was quantified using the master fault trees, database, and other supporting files developed for the updated DBNPS PSA. Quantification was performed using CAFTA version 3.2b and PRAQUANT. The DBNPS PSA model was recently updated to reflect the as-built, as-operated plant. Additionally, subsequent to the update a configuration control process was put in place to maintain the PSA as close as possible to actual plant configuration. In November 1999, the PSA was reviewed as the pilot plant for the Babcock and Wilcox (B&W) Owners Group peer review process.

The boric acid dilution fault tree was quantified with the initiating event frequency set to 1.0 and the truncation set to 1.0E-6. After the cut sets were generated, the initiating event frequency was set to 1.1E-5 giving an effective truncation of 1.1E-11. The quantification resulted in a CDF of 1.1E-7 per year.

Evaluation of Post Modification CDF with Additional Line from HPI Pump 1-1

To evaluate the impact of single failure on the results a second case was analyzed. For this calculation it was assumed that the boric acid dilution modification included a line from HPI Pump 1-1 to the auxiliary spray line. This would allow HPI Pump 1-1 to provide flow to the auxiliary spray line with suction provided by DH/LPI Pump 1-1. If DH/LPI Pump 1-1 is not available it would be possible to cross connect the LPI discharge to provide flow from DH/LPI Pump 2.

The fault tree logic was revised to reflect the additional BPC path but otherwise the analysis was performed using a method similar to that described above for the single HPI BPC line. The quantification resulted in a CDF of 1.3E-8 per year.

Evaluation of Large Early Release Frequency (LERF)

The calculation of LERF was somewhat complicated because the plant damage states for failure of long term boric acid dilution sequences did not fit in with any plant damage states previously calculated for large or medium LOCAs. Therefore, the LERF solution involved identifying new plant damage states, solving the plant damage states, creating new containment event tree flag files, solving the CET and combining the results.

The bridge trees developed for the updated DBNPS PSA were reviewed to determine applicable plant damage states for failure of boric acid dilution. The success of several events in the bridge trees is implied by the boric acid dilution sequences. The events with implied success are AC power available, RCS depressurization, borated water storage tank (BWST) injection and heat removal by low pressure recirculation. The five plant damage states listed in Table 4 were determined to apply for failure of boric acid dilution sequences.

Table 4. Composition of Plant Damage States

Plant Damage State		Containment (CTMT) System (CS) States						
Core Damage Bin	**CS State**	**AC Power Available**	**RCS Status**	**CTMT Isolated**	**CTMT Heat Removal Available**	**CTMT Spray Available**	**RCS Depress.**	**Borated Water Storage Tank Status**
LBD	01Y	Yes	LOCA > seal LOCA	Yes	Yes	Yes	Yes	Injected before vessel breach
	04Y	Yes	LOCA > seal LOCA	Yes	Yes	No	Yes	Injected before vessel breach
	01S	Yes	LOCA > seal LOCA	Small Isolation Failure	Yes	Yes	Yes	Injected before vessel breach
	04S	Yes	LOCA > seal LOCA	Small Isolation Failure	Yes	No	Yes	Injected before vessel breach
	01L	Yes	LOCA > seal LOCA	Large Isolation Failure	NA	NA	Yes	Injected before vessel breach

Fault trees corresponding to the sequences of interest were developed. Additional logic was appended to the core damage bins to account for the relevant failures in the bridge tree. Success states are removed based on bridge tree logic as is done for core damage sequences. Very low truncations were used in the plant damage states quantification to ensure an adequate number of cut sets were generated for each sequence. The truncations used are listed in Table 5. Table 6 lists the quantification results.

Table 5. Level 2 Sequence Truncation

Sequence	PRAQUANT Truncation	Effective Truncation
LBD_01Y	1.0E-6	1.1E-11
LBD_04Y	1.0E-8	1.1E-13
LBD_01S	1.0E-10	1.1E-15
LBD_04S	1.0E-10	1.1E-15
LBD_01L	1.0E-10	1.1E-15

Table 6. Plant Damage State Frequencies

Sequence	Core Damage Frequency
LBD_01Y	1.1E-7
LBD_04Y	1.9E-9
LBD_01S	9.4E-12
LBD_04S	1.2E-13
LBD_01L	3.1E-12

The containment event tree, as described for the updated PSA level 2 analysis, was quantified to calculate the containment response for each of the plant damage states. The flag files for core damage due to boric acid precipitation are similar to those for LOCAs. However, some events were modified since core damage due to boric acid precipitation implies the success of low pressure recirculation. The containment event tree was quantified using GTPROB32.

To calculate the large early release frequency the plant damage state frequencies were multiplied by each of the CET end state probabilities and the end states that contribute to LERF were summed. Table 7 summarizes the large early release and containment failure mode frequencies calculated for failure of boric acid dilution sequences.

Table 7. Containment Failure Mode Frequencies

Failure Mode	Frequency
Large Early Release	1.1E-11
Early Containment Failure	4.5E-10
Containment Sidewall Failure	0.0*
Containment Bypass Failure	0.0*
Late Containment Failure	3.0E-12
Containment Basemat Failure	1.1E-09
Small Isolation Failure	9.5E-12
Large Isolation Failure	3.1E-12
No Containment Failure	1.1E-7

* These outcomes are logically precluded for the initiating events relevant to this assessment.

Evaluation of LERF with Additional Line from HPI Pump 1-1

The fault tree logic was revised to reflect the additional BPC path but otherwise the analysis was performed using a method similar to that described above for the single HPI BPC line. The quantification resulted in a LERF of 2.6E-12 per year.

Results of the Level 1 Analysis

Examination of the cutsets derived from analysis of the post modification condition reveals that ECCS room cooler failures contribute about 65% of the calculated CDF. Failure by the operators to align BPC comprises the next largest portion of the CDF. Single electrical component failures are a small contributor due to the lower probabilities of these events coincident with a large LOCA.

The addition of the second HPI line resulted in cutsets dominated by human failures. The common event representing the failure to align BPC by either active method (ZHAC2H4E) was a significant contributor in this case.

Results of the Level 2 Analysis

The LERF was calculated to be very small for either post modification configuration. This very small LERF contribution was expected for two reasons. Generally the LERF contribution from accidents that result in reactor coolant system depressurization is small because induced tube ruptures and core debris dispersal beyond the cavity are both precluded at low RCS pressure. Additionally, for failure of boric acid dilution sequences, the success of low pressure injection and low pressure recirculation is implied. Therefore, at a minimum, the BWST would be injected and heat removal by low pressure recirculation would be available.

References

1. DBNPS Calculation C-NSA-099.16-26, *Long Term Boron Dilution Modification 97-0074*, Revision 0, January 2000.
2. *Pipe Failure Study Update*, EPRI TR-102266, April 1993.
3. FTI Calculation 86-5006059-00, *Post-LOCA Boron Management for Davis-Besse,* March 2000.

Summary of Decay Heat Removal Cooler Evaluation

Background

As a part of the supporting analysis of the new backup BPC methodology, it was determined that while the boric acid concentration in the core was shown to stay well below the solubility limit for the fluid temperatures involved, the boric acid concentration in the DHR drop line coming from the core could exceed the solubility limit when the solution reached the DHR cooler, due to the cooling effect of the cooler. Therefore, additional analyses were performed, using more realistic assumptions (described below), to demonstrate that boric acid precipitation in the DHR cooler will not occur.

Assumptions

The DHR cooler performance analysis conservatively assumed full flow through the cooler. The capability exists to minimize flow through the DHR cooler by opening the DHR cooler bypass valve (DH-13A or DH-13B, depending on which DHR train is being used for the BPC function) and closing the DHR cooler outlet valve (DH-14A or DH-14B). However, due to the DHR pump recirculation line, flow through the cooler cannot be completely bypassed. The DHR pump recirculation line connection is downstream of the outlet of the cooler, and upstream of the cooler outlet valve, and cannot be isolated by the control room operators. Minimization of flow through the DHR cooler would limit the amount of concentrated boric acid solution that would be exposed to the cold temperatures, reducing the quantity of precipitate should the solubility limit be reached, and reducing the likelihood of plugging the cooler tubes.

The core concentration analysis assumed an initial boric acid concentration of 2400 ppm in the core. In that the boric acid concentration rapidly drops during the first few days of operation after refueling, this is a very conservative assumption. Within several Effective Full Power Days (EFPD) of operation, the concentration would decrease to approximately 1700 to 1800 ppm. Therefore, a more realistic concentration of 1900 ppm was assumed for the detailed analysis of the DHR cooler performance. The reduced initial boric acid concentration allows more time before solubility limits would be reached. As described below, although this assumption results in less conservatism during the first several EFPD of operation, the effects are somewhat offset by the over-conservatism applied to the assumed decay heat level during this same time period.

The amount of decay heat available to cause boric acid concentration was also very conservative in the core concentration analysis. A decay heat level of 1.2 times the ANS Standard (Reference 1) infinite operation decay heat was utilized throughout the core concentration analysis. A more realistic decay heat level of 1.0 times the ANS Standard infinite operation decay heat was utilized in the DHR cooler performance analysis, reducing the boric acid concentration rate in the core. It should be noted that since this decay heat level assumes infinite operation, it includes considerable margin. The margin of decay heat one hour or more after shutdown is about 20 percent during the first 20 EFPD of operation. Therefore, this additional margin early in core life offsets the above-mentioned lower initial boric acid concentration assumed during the first several EFPD of operation.

The boron solubility limit used in the core concentration analysis was based on the boric acid weight fraction reduced by 4 weight percent boric acid. This weight fraction was used over the entire temperature range, rather than taking a percentage of the solubility limit at any given temperature. This is conservative for the temperatures in the core. However, the impact of this margin grows as the temperature is reduced, since the solubility limit is shrinking, while the margin remains fixed. At 104 °F, the margin term is half as large as the actual solubility limit. Therefore, for the DHR cooler analysis, in order to reduce the dominance of the margin on the results, a more appropriate acceptance limit was utilized, consisting of the boric acid concentration corresponding to 90 percent of the solubility limit. This ensures that a margin is maintained, yet provides a more realistic estimate of the approach to the solubility limit.

The DHR cooler analysis assumed a boric acid solution temperature of 95 °F in the DHR cooler, corresponding to the normal control temperature of the component cooling water (CCW) inlet flow to the DHR cooler. Additional conservatism is present in that there is actually a temperature differential across the walls of the tubes of the DHR cooler, such that the boric acid solution will actually be above the temperature of the CCW inlet. Since the solubility curve changes rapidly as a function of temperature in this temperature range, the margin introduced by this conservatism is potentially significant. The analysis assumption is conservative in that the minimum expected CCW inlet temperature will most likely exceed 95 °F post-LOCA, due to the increased heat loads on the CCW system. However, since service water (SW) flow to the CCW heat exchangers is maximized post-LOCA, the CCW inlet temperature to the DHR cooler could be less than 95 °F, depending on the SW temperature. Therefore procedural controls will be established to ensure that the CCW inlet flow to the DHR cooler is at least 95 °F prior to establishing the backup BPC method. The equipment necessary to perform any required warm up of the CCW system will be accessible following the LOCA, and sufficient time exists to perform the warm up.

<u>Result of Evaluation</u>

A preliminary evaluation demonstrated that, based on the above assumptions, the boric acid concentration of the solution being introduced into the DHR cooler was significantly lower than previously analyzed. However, the concentration was still above the solubility limit of the 95 °F temperature assumed for the cooler.

Consequently, it was necessary to credit flow through the reactor vessel outlet nozzle gaps, but only for the time period until the backup BPC method would be established. The analysis demonstrates that with the differential temperature existing between the reactor vessel internals and the reactor vessel shell, the gaps will be of sufficient size to limit the concentration to below the cooler solubility limit. Further discussion regarding the size of the gaps and the potential for debris to clog the gaps is provided below.

Reactor Vessel Outlet Nozzle Gap Size

The reactor vessel outlet nozzle gaps exist as a result of the construction of the reactor vessel and its internals. In order to assemble and disassemble the internals for reactor vessel inspection, a small gap is required between the hot leg nozzles and the upper plenum assembly. When the temperature of the reactor vessel internals is less than or equal to the temperature of the reactor vessel shell, the gaps will be open.

During a LOCA, the reactor vessel shell cools slower than the internals, which rapidly respond to and remain in equilibrium with the temperature of the water in the core region. This temperature differential will result in a gap size that is proportional to the temperature difference between the internals and the reactor vessel. Credit for gap flow is not taken until the upper plenum refills with liquid and a differential temperature sufficient to open the gaps is predicted. The need for gap flow credit ends with the start of dilution flow. The temperature difference at that time is still above the minimum to open the gaps. Therefore, sufficient flow area will exist to support the conclusion that gap flow will limit the core boric acid concentration to below the cooler solubility limit.

Effect of Debris on the Reactor Vessel Outlet Nozzle Gaps

If the reactor vessel outlet nozzle gaps became clogged with debris, the flow area could be reduced below that needed to support gap flow dilution. One potential source of debris is the containment emergency sump. However, this source of debris is not of concern since an active BPC method would be established immediately after switchover from the BWST to the ECCS emergency sump. Upon establishment of an active BPC method, gap flow is no longer required to be credited.

Another potential source of debris is that present in the RCS during steady state operations and following transients, including reactor trips. A study of RCS particulate matter in Belgian Pressurized Water Reactors (PWRs) was reviewed (Reference 2). The study included an evaluation of RCS particle mass density during steady state operations and following transients, including reactor trips. The study determined that the largest particle size is less than 30 microns in size. The gap width for which boric acid dilution flow was credited is between 0.01 inches (254 microns) and 0.25 inches (6350 microns) in size. Therefore, it is unlikely that any particulate matter would obstruct gap flow.

Another potential source of debris is that generated in the RCS by fuel pin rupture in the core, as could occur during the violent blowdown and reflood phases of the LOCA. LOCA analyses have shown that LOCAs smaller than 0.75 ft^2 will not result in fuel clad rupture. Fuel pin ruptures are more likely to occur during the lower plenum refill period, while the fuel is undergoing nearly adiabatic heatup. Fragments from ruptures that occur during this period would likely fall downward in the fuel assemblies, since there is little upward flow through the core. Smaller fragments would fall to the lower fuel assembly debris filter or the lower plenum, but larger fragments could be trapped within the fuel pin lattice. As the water level builds in the core region, significant quenching and steam generation will occur that could dislodge the particles and entrain them. The fuel grid lattices above the particles would tend to capture the larger particles, but smaller sized fragments and any particles that breakup due to the up flow would be carried up and out of the core, into the upper plenum of the vessel.

Since the upper plenum is initially steam filled, the fuel fragments will detrain from the flow, as they enter the larger area and flow velocities drop rapidly. Test results (Reference 3) indicate that approximately half of the water droplets entering the upper plenum detrain and pool or form a film on metal structures at the beginning of the reflood phase in the core. A similar or larger fraction of the fuel fragments would also be expected to detrain since the fragment density is much greater than the liquid that is dropping out of the flow stream. A small fraction of the smallest particles may be carried out the Reactor Vessel Vent Valves (RVVVs) with the steam and water droplet mixture that is passing through the valves. From there, the fragments could enter the downcomer and would likely not be returned to the upper plenum.

Upward flow of fuel fragments would be expected to subside significantly after the quench front passes through the core and steady boiling commences. The flow velocities will no longer be great enough to lift the particles into the upper plenum. As the two phase mixture refills the upper plenum, the smaller fragments in that region could be re-entrained by the bubbly flow. The particles would remain agitated until the level is sufficient to support flow through the plenum cylinder holes into the outlet annulus. The larger particles that reach the outlet annulus region could settle to the bottom and remain there, removed from the vessel two-phase recirculation flows. Smaller particles that remain in suspension could be transported to the entrance of the hot leg gaps by the upper plenum recirculation flows. Any particles near the entrance to the gaps that are smaller than the gaps will pass through without obstructing the flow path. While larger particles could plug the gaps, it has been determined that very few particles of larger size could reach the gap region. The largest fuel fragments expected to reach the gap area are 3600 microns in diameter. The gap width at the earliest time that the largest particles could be transported to the gap region (approximately 150 seconds after a large break LOCA) is already 3600 microns wide and increasing. After 150 seconds, the highest steam and liquid droplet velocities that could sweep the larger fuel fragments into the upper plenum would begin a slow but steady decline. The gaps will continue to open as the internals cool until they reach an approximate upper size of 0.25 inches (6350 microns). This width will allow any particles that initially might partially obstruct the opening to be swept on through, so that full gap flow will be obtained. After the internals have cooled to near the RCS saturation temperature, the gap size will begin to decrease due to the slow, but steady, cooldown of the reactor vessel shell. The gap size will remain above the maximum particle size that could pass through the spacer grids of the fuel for at least 45 minutes, based on the most conservative estimates. Because credit for gap flow has only been taken until the backup BPC method is initiated immediately following transfer to the emergency sump, the gap sizes are large enough that the fuel fragments will not obstruct the dilution flow through the gaps.

Summary

The amount of flow calculated to occur through the hot leg nozzle gaps is sufficient to keep the boric acid concentrations below the solubility limits of the DHR coolers when more realistic, but still conservative, core conditions are used in the analysis.

Investigation of the types and sizes of debris within the RCS immediately following a LOCA has shown that particulate matter that could be postulated to reach the entrance of the gaps is not of sufficient size to obstruct the flow through the gaps. Additionally, gap plugging from fuel debris due to large break LOCA-induced fuel pin ruptures was considered, and it was concluded that particulates would not obstruct the gap flow before the backup BPC flow path would be established. The fuel fragments are potentially of sufficient size to block the gaps several hours after the LOCA, however, they are unlikely to reach the gaps at this time because the velocities necessary to carry the large particles will not exist.

Based on this investigation, it is concluded that flow through the hot leg nozzle gaps will occur and will adequately limit the concentration of boric acid in the water that would initially enter the DHR cooler, so that it will not experience precipitation. Once flow via the backup method is initiated, the concentration entering the DHR cooler will be diluted quickly by ECCS Emergency Sump water, and no credit for flow through the gaps is needed to show protection against precipitation in the core region or in the DHR coolers.

References

1. American Nuclear Society (ANS) Standard 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fuel Thermal Reactors (DRAFT)," October 1971.

2. AEA Technology, "The Nature and Behavior of Particulates in PWR Primary Coolant," EPRI NP-6640, December 1989.

3. MPR Associates, Inc. Report Prepared for the USNRC, "Summary of Results from the UPTF Carryover/Steam Binding Separate Effects Tests, Comparison to Previous Scaled Tests, and Application to U.S. Pressurized Water Reactors," MPR-1213, October, 1990.

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 (419-321-8466) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	**DUE DATE**
1. Prior to initiation of the primary BPC method, the operator will confirm the need for BPC based on RCS conditions.	1. During implementation of the planned plant modification.
2. Specific guidance will be provided to control room operators to ensure that no vapor entrainment will damage the DHR/LPI pump. [For the backup BPC method].	2. During implementation of the planned plant modification.
3. Valves DH-11 and DH-12 will be opened after sump switchover, provided the RCS is within the design pressure and temperature range for the DHR drop line piping and components.	3. During implementation of the planned plant modification.
4. Procedural controls will be in place to alert operators to the need for establishing BPC, and to ensure that they respond promptly to restore the BPC function as quickly as possible if a failure occurs that precludes using both [primary and backup] methods.	4. During implementation of the planned plant modification.
5. Procedural controls will be established to ensure that the CCW inlet flow to the DHR cooler is at least 95 °F prior to establishing the backup BPC method.	5. During implementation of the planned plant modification.