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Docket Number 50-346

License Number NPF-3

Serial Number 2667

November 9, 2000

United States Nuclear Regulatory Commission
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Washington, DC 20555-0001

Subject: License Amendment Application to Revise Technical Specifications to Extend the Allowed Outage Time for One Low Pressure Injection System/Decay Heat Cooler Train and Containment Spray System Train (License Amendment Request No. 98-0004, TAC No. MA9694)

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specifications. The proposed changes involve: Technical Specification (TS) TS 3/4.5.2, "Emergency Core Cooling Systems - ECCS Subsystems – $T_{avg} \geq 280°F$;" TS 3/4.6.2.1, "Containment Systems – Depressurization and Cooling Systems – Containment Spray System;" and TS Bases Section 3/4.5.2 and 3/4.5.3, "ECCS Subsystems."

This License Amendment Request proposes a revision of the Action statements of the DBNPS TS Limiting Condition for Operation (LCO) 3.5.2 and 3.6.2.1. This request proposes the extension of the allowed outage time for one Low Pressure Injection (LPI) System/Decay Heat Cooler train of an Emergency Core Cooling System (ECCS) subsystem from 72 hours to 7 days (168 hours) for LCO 3.5.2. One Containment Spray System train may be impacted by the inoperability of the associated LPI train. Therefore, this License Amendment Request also proposes the extension of the allowed outage time for one train of the Containment Spray System from 72 hours to 7 days for LCO 3.6.2.1. Additional information is proposed to be added to TS Bases Section 3/4.5.2 and 3/4.5.3 to clarify the TS LCO 3.5.2 requirements.

In the Babcock & Wilcox plant design for plants such as the DBNPS, the Decay Heat Removal (DHR) and LPI systems are combined. The proposed allowed outage time will allow

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meaningful DHR/LPI System train maintenance to be performed with the plant at power. The maintenance to be performed at power includes both corrective maintenance, and voluntary entry into the LCO for preventative maintenance and testing. Preventative maintenance, now performed during a plant refueling outage on a DHR/LPI System train, will be able to be performed with the plant at power, thus removing this maintenance activity from the refueling outage (when the DHR System is in service).

These proposed changes are based on the Babcock & Wilcox Owners Group (B&WOG) Topical Report BAW-2295A, Revisions 1 & 2, "Justification for the Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems," that was submitted to the NRC by letter dated October 9, 1998. This Topical Report addressed seven B&W plant units, including the DBNPS. The DBNPS is the lead B&W plant unit requesting these changes to be made to its Technical Specifications. The NRC staff evaluated Topical Report BAW-2295A, Revision 1 and concluded in its Safety Evaluation (TAC No. MA3807; dated June 30, 1999), that the proposed allowable outage time of 7 days was acceptable. Revision 2 of the Topical Report incorporates the NRC's Safety Evaluation and the B&WOG responses provided to NRC requests for additional information during the NRC staff's review of the Topical Report.

The DBNPS requests that this License Amendment Request be approved by the NRC and the amendment issued by June 15, 2001, so that it may be utilized during the current operating cycle.

Should you have any questions or require additional information regarding this application, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

FWK/laj

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, DB-1 NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
K. S. Zellers, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. Also included is the Safety Assessment and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial Number 2667) concern Appendix A, Technical Specifications (TS):

TS 3/4.5.2 "Emergency Core Cooling Systems - ECCS Subsystems – $T_{avg} \geq 280°F$"

TS 3/4.6.2.1 "Containment Systems – Depressurization and Cooling Systems – Containment Spray System."

TS Bases Section 3/4.5.2 and 3/4.5.3, "ECCS Subsystems."

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: _____
Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 9th day of November, 2000.

_____
Notary Public, State of Ohio - Nora L. Flood
My Commission expires September 4, 2002.

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Facility Operating License Number NPF-3, Appendix A, Technical Specifications. The proposed changes involve Technical Specification TS 3/4.5.2 "Emergency Core Cooling Systems - ECCS Subsystems – $T_{avg} \geq$ 280°F" and TS 3/4.6.2.1 "Containment Systems – Depressurization and Cooling Systems – Containment Spray System."

A. Time Required to Implement: The License Amendment associated with this license amendment application is to be implemented within 120 days after NRC issuance.

B. Reason for Change (License Amendment Request Number 98-0004):

This License Amendment Request proposes a revision of the Action statements of the DBNPS TS Limiting Condition for Operation (LCO) 3.5.2 and 3.6.2.1. This request proposes the extension of the allowed outage time for one LPI System/Decay Heat Cooler train of an ECCS subsystem from 72 hours to 7 days (168 hours) for LCO 3.5.2. One Containment Spray System train may be impacted by the inoperability of the associated LPI train. Therefore, this License Amendment Request also proposes the extension of the allowed outage time for one train of the Containment Spray System from 72 hours to 7days for LCO 3.6.2.1.

In the Babcock & Wilcox plant design for plants such as the DBNPS, the Decay Heat Removal (DHR) and LPI systems are combined. The proposed allowed outage time will allow meaningful DHR/LPI System train maintenance to be performed with the plant at power. The maintenance to be performed at power includes both corrective maintenance, and voluntary entry into the LCO for preventative maintenance and testing. Preventative maintenance, now performed during a plant refueling outage on a DHR/LPI System train, will be able to be performed with the plant at power, thus removing this maintenance activity from the refueling outage (when the DHR System is in service).

C. Safety Assessment and Significant Hazards Consideration: See Attachment.

Docket Number 50-346
License Number NPF-3
Serial Number 2667
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NO. 98-0004

(25 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 98-0004

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, Facility Operating License NPF-3, Appendix A Technical Specifications (TS) to Revise TS 3/4.5.2, "Emergency Core Cooling Systems - ECCS Subsystems – $T_{avg} \geq 280°F$," TS 3/4.6.2.1, "Containment Systems – Depressurization and Cooling Systems – Containment Spray System," and TS Bases Section 3/4.5.2 and 3/4.5.3 "ECCS Subsystems"

DESCRIPTION:

At the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, the Emergency Core Cooling System (ECCS) consists of Core Flooding Tanks, a High Pressure Injection (HPI) System and a Low Pressure Injection (LPI) System (including decay heat coolers). The HPI System consists of two separate, redundant HPI trains. Each HPI train consists of a HPI pump and associated piping, valves, instrumentation and controls. The HPI pumps will take emergency coolant from the Borated Water Storage Tank (BWST) and deliver it to the reactor through the cold legs of the Reactor Coolant System.

The LPI System consists of two separate, redundant LPI trains. Each LPI train consists of a LPI pump and associated piping, valves, instrumentation and controls. LPI pumps will take emergency coolant from the BWST, and following the BWST's depletion, from the Containment Emergency Sump. The LPI piping provides flow to the reactor vessel through the core flooding nozzles located in the reactor vessel.

Technical Specification (TS) requirements for the HPI System and LPI System for Modes 1 (power operation), 2 (startup), and 3 (hot standby) are stated in TS 3/4.5.2, "ECCS Subsystems – $T_{avg} \geq 280°F$." The TS requirements for the Core Flooding Tanks are in TS 3/4.5.1, "Core Flooding Tanks," and are not affected by the proposed changes.

The major components of the DBNPS Containment (Reactor Building) Spray System consist of two trains of a Containment Spray Pump, associated piping, valves, instrumentation, and controls. Each Containment Spray System train shares some common suction piping with its corresponding LPI System train. The Containment Spray System Pumps will take emergency coolant from the BWST, and following the BWST's depletion, from the Containment Emergency Sump. Technical Specification requirements for the Containment Spray System for Modes 1, 2, 3 and 4 (hot shutdown) are stated in TS 3/4.6.2.1, "Containment Spray System."

This License Amendment Request proposes a revision of the Action statements of the DBNPS TS Limiting Condition for Operation (LCO) 3.5.2 and 3.6.2.1. This request proposes the extension of the allowed outage time for one LPI pump/decay heat cooler train of the LPI System from 72 hours to 7 days (168 hours) for LCO 3.5.2. One Containment Spray train may be impacted by the inoperability of the associated LPI train. Therefore, this License Amendment Request also proposes the extension of the allowed outage time for one train of the Containment Spray System from 72 hours to 7days for LCO 3.6.2.1. In addition, a discussion regarding the operability requirements for the HPI and LPI trains is proposed to be added to TS Bases Section 3/4.5.2 and 3/4.5.3, "ECCS Subsystems."

In the Babcock & Wilcox plant design for plants such as the DBNPS, the Decay Heat Removal (DHR) and LPI systems are combined. The DHR System provides cooling for the reactor core when the plant is shutdown. The proposed allowed outage time will allow meaningful DHR/LPI System train maintenance to be performed with the plant at power. The maintenance to be performed at power includes both corrective maintenance, and voluntary entry into the LCO for preventive maintenance and testing. Preventative maintenance, now performed during a plant refueling outage on a DHR/LPI System train, will be able to be performed with the plant at power thus removing this maintenance activity from the refueling outage (when the DHR System is in service).

The current ECCS LCO 3.5.2 Action statement reads as follows:

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

b. 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.

It is proposed that, in addition to increasing the allowed outage time from 72 hours to 7 days, this Action statement be revised to provide separate Actions for the HPI System trains and the LPI System trains with their associated decay heat coolers. Separate actions are proposed in order to clearly identify the allowed outage time for a LPI train and associated decay heat cooler, and a HPI train.

The proposed revised LCO 3.5.2 Action statement would read as follows:

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.

The current LCO 3.6.2.1 ACTION Statement reads as follows:

> With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

The proposed revised LCO 3.6.2.1 ACTION statement would read as follows:

> With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

These proposed changes are based on the Babcock & Wilcox Owners Group (B&WOG) Topical Report BAW-2295A, Revisions 1 & 2, "Justification for the Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems," that was submitted to the NRC by letter dated October 9, 1998. This Topical Report addressed seven B&W plant units, including the DBNPS. The DBNPS is the lead B&W plant unit requesting these changes to be made to its Technical Specifications. The NRC staff evaluated Topical Report BAW-2295A, Revision 1 and concluded in its Safety Evaluation (TAC No. MA3807; dated June 30, 1999), that the proposed allowable outage time of 7 days was acceptable. Revision 2 of the Topical Report incorporates the NRC's Safety Evaluation and the B&WOG responses provided to NRC requests for additional information during the NRC staff's review of the Topical Report.

A discussion is also proposed to be added to TS Bases Section 3/4.5.2 and 3/4.5.3 as follows:

> 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.

This discussion is being added to explain the makeup of an ECCS subsystem.

Discussion would also be added to these TS Bases to address the flexibility in plant operation provided by the LCO and Actions as follows:

> 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.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The systems, components, and activities affected by the proposed changes are the TS allowed outage times for the Low Pressure Injection System/decay heat cooler and the Containment Spray System. The increased allowed outage time will allow additional pump and valve maintenance and testing to be performed with the plant on-line (i.e., at power).

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

<u>High Pressure Injection System and Low Pressure Injection System</u>

The ECCS is described in the DBNPS Updated Safety Analysis Report (USAR) Section 6.3., "Emergency Core Cooling System." It is made up of the High Pressure Injection System, the Low Pressure Injection System, and the Core Flooding System. The High Pressure Injection System provides pumped injection of water into the Reactor Coolant System (RCS) cold leg piping. The HPI System is used to provide cooling water to the reactor core when the Low Pressure Injection System is unable to, which could occur as a result of a Small Break Loss of Coolant Accident in the RCS with the RCS pressure greater than the LPI pump head. The Core Flooding System, made up of two pressurized storage tanks, provides water at medium to low pressure directly to the reactor vessel downcomer through a set of nozzles. The system is short term and ceases to operate when the tanks empty. It is not affected by this License Amendment Request. The Low Pressure Injection System provides pumped injection of low pressure water directly into the reactor vessel downcomer through the same nozzles as the Core Flooding System. The

system is used for long term cooling of the reactor core during a Loss-of-Coolant Accident (LOCA).

The ECCS provides the capability to meet the functional requirements over both the short and long term duration of the LOCA. Following a LOCA, assuming a simultaneous loss of normal power source, the HPI and LPI Systems are powered by the Emergency Diesel Generators and will operate with no loss of function to pump water from the Borated Water Storage Tank into the Reactor Coolant System to maintain core cooling. Separate and independent flow paths are provided in the ECCS, and redundancy in the active components ensures that the required functions will be performed if a single active failure occurs. Separate essential power sources are supplied to the redundant active components, and separate Safety Features Actuation System (SFAS) instrumentation channels are used to actuate the system. During the injection phase, the HPI System and LPI System will operate to provide full protection over the entire spectrum of break sizes. The HPI System will prevent any significant uncovering of the core in the event that small coolant piping leaks occur when RCS pressure is maintained above 600 psig and below 1500 psig and will delay uncovering of the core when intermediate-size leaks occur. The Core Flooding and LPI Systems will inject borated water into the core at intermediate to low RCS pressures and will ensure adequate core cooling for break sizes ranging from intermediate to the double-ended rupture of the RCS piping in either the hot or cold leg. During the recirculation phase, the LPI System will recirculate the spilled reactor coolant and injection water from the Containment Emergency Sump to the reactor vessel and will maintain long-term core cooling.

For small breaks, the RCS pressure may be higher than the maximum LPI pump head at the time of Containment Emergency Sump water recirculation. Under these circumstances crossover connections permit alignment of the HPI pumps to take suction from the outlet of the LPI pumps' Decay Heat Coolers to provide for Containment Emergency Sump water recirculation to the reactor core. The valves are motor operated with handswitches in the control room. This mode of operation, whereby the LPI pumps and HPI pumps are combined is called "piggy-backing."

The LPI System, including the decay heat coolers, also has non-emergency functions. In the Decay Heat Removal (DHR) System mode, it is used to remove decay heat during shutdown and refueling operations. The DHR System takes suction from the RCS hot leg and returns it back to the RCS after passing it through a DHR cooler in each train. In addition, the DHR System may be configured for use as a backup system for Spent Fuel Pool cooling.

Containment Spray System

The Containment Spray System is described in the DBNPS USAR Section 6.2.2.2.2, "Containment Spray System." It is an engineered safety feature which has the dual function of removing heat and fission product iodine from the post-accident containment atmosphere. Removal of heat is accomplished by directing borated water spray into the

containment atmosphere. The absorption of iodine by the containment spray is accomplished mostly due to the large amount of surface area continuously available for mass transfer between the spray solution and the containment atmosphere. The removal of airborne iodine from the containment atmosphere serves to limit the potential dose to receptors located at the site boundary and outer boundary of the low population zone to within 10 CFR 100 guideline values.

The Containment Spray System serves no function during normal operation. The system consists of two half-capacity pumps, two half-capacity spray headers, isolation valves, and the necessary piping, instrumentation, and controls. The pumps and valves can be remote manually operated from the control room. High Containment Vessel atmosphere pressure or low RCS pressure will actuate a SFAS trip to open the spray isolation valves. "High-high" containment atmosphere pressure will actuate a SFAS trip to start two containment spray pumps. Like the LPI and the HPI Systems, the pumps take suction initially from the Borated Water Storage Tank (BWST). The Containment Spray System shares the BWST and some of the suction lines from the BWST with the HPI and LPI Systems. After the water in the BWST reaches a low level, the spray pump suction is transferred to the Containment Emergency Sump. A BWST low level alarm and permissive alarm is set above the bottom of the BWST to alert the operator in sufficient time to perform the switchover without interrupting spray operation.

EFFECTS ON SAFETY:

Increase In Allowable Outage Times

The Framatome Technologies Topical Report BAW-2295A, Revisions 1 and 2 (Reference 1) provides the technical basis for the proposed change to the Action statements of the DBNPS TS 3/4.5.2 Emergency Core Cooling Systems – ECCS Subsystems – $T_{avg} \geq$ 280°F and TS 3/4.6.2.1, Containment Systems – Depressurization and Cooling Systems – Containment Spray System. The Topical Report provided the deterministic evaluation and risk assessment to support the increase in the TS allowable outage time for one train of inoperable LPI and Containment Spray from 72 hours to 7 days. The Topical Report used plant-specific probabilistic risk assessment (PRA) to assess the risk impact of increased Low Pressure Injection System and Containment Spray System unavailability. The NRC staff evaluated this Topical Report and found the proposed increase in allowed outage time acceptable in its Safety Evaluation. A summary of the Topical Report and the NRC staff's evaluation follows below.

Deterministic Evaluation:

The Topical Report's deterministic evaluation consisted of a review of plant systems and safety functions impacted by entry into the LPI or Containment Spray System TS allowable outage times. The affected DHR/LPI or Containment Spray System safety functions were quantitatively and qualitatively assessed. It was determined that no new accidents or transients would be introduced by the proposed changes.

The impact on the proposed changes on the safety margins was also considered. Extending the allowable outage time to 7 days for one inoperable train does not impact any assumptions or inputs in the USAR. The NRC found the deterministic evaluation acceptable.

Probabilistic Risk Assessment:

The Topical Report's probabilistic risk assessment used to assess the impact of the proposed changes is based upon similar measures defined in Regulatory Guide (RG) 1.174, "An approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications." The risk impacts of the proposed changes were calculated and compared against the acceptability guidelines in the RGs of 1.0E-4 for Core Damage Frequency (CDF) and 1.0E-5 for Large Early Release Frequency (LERF). The DBNPS CDF for the proposed change was 1.76 E-5 (overall CDF using the full 168 hours allowed outage time per year), and the DBNPS LERF for the proposed change was less than or equal to 3.3E-7 (LERF using the full 168 hours allowed outage time per year). The RG 1.177 incremental conditional core damage probability (ICCDP) value of 5E-7 was slightly less than those calculated for the B&W plants (the DBNPS value was 1.5E-6). The NRC Safety Evaluation, however, found the ICCDP values acceptable due to the following compensatory measures that lower the risk impacts:

- Avoiding simultaneous outages of additional risk-significant components during the allowable outage times of the LPI and Containment Spray System trains. These components include both Auxiliary Feedwater System trains, both HPI trains (for reasons other than inoperable due to the associated LPI train), both Containment Air Cooler trains, and their power supplies.
- Defining specific criteria for scheduling only those preventive maintenances that can be completed within the 7 day allowable outage time.
- Assuring that the frequency of entry into the allowable outage time and consequently, the average maintenance duration per year remains within that assumed in the Topical Report.
- Taking measures to assure that when maintaining the LPI and Containment Spray System trains, both are not made unavailable unless it is necessary.

The DBNPS will ensure these compensatory measures are taken for outages of the LPI and Containment Spray Systems. The implementation of the DBNPS configuration risk management processes regarding the use of the extended allowable outage time for the LPI and Containment Spray Systems is addressed in the next section. Also addressed, as required by the NRC staff's Safety Evaluation, are updates of the DBNPS probabilistic risk assessment and information on the DBNPS probabilistic risk assessment quality.

Configuration Risk Management Program (CRMP):

The DBNPS has in place a CRMP that determines the risk associated with the removal of equipment from service and prohibits entry into extended LPI System and /or Containment Spray System allowable outage times for scheduled maintenance purposes if external event conditions or warnings are in effect, and establishes minimum compensatory actions based on the risk level. This program is designed to prevent the simultaneous outages of risk significant components and reduce the potential of high risk configurations. It is used to assess the risk associated with the removal of equipment from service during extended LPI System and Containment Spray System allowable outage times. The guidance for the CRMP is contained in the administratively-controlled Administrative Work Process Guidelines Manual, Work Scheduling Guideline, WPG-1. The CRMP that has been implemented per WPG-1 contains the four key elements described in Section 2.3, "Configuration Risk Management Program," of the NRC's Safety Evaluation for Topical Report BAW-2295A. These four key CRMP elements, as summarized, include:

- Implementation of the Maintenance Rule (10CFR50.65(a)(3))
- Monitoring, assessing and dispositioning plant modifications and procedure changes as part of the normal probabilistic risk assessment update process.
- Basing the CRMP on a Level 1, at power, internal events probabilistic risk assessment model that may use any combination of quantitative and qualitative input.
- Treating external events and Level 2 issues qualitatively and/or quantitatively.

Additionally, the WPG-1 presently includes the following elements:

- A copy of the proposed weekly work schedule is provided to the Nuclear Engineering Department for an assessment of overall risk. The Nuclear Engineering Department provides an overall risk assessment and any recommended changes to the Operations Work Control department.
- The risk assessment submitted by Nuclear Engineering provides a daily risk summary that ranks maximum risk in one of four risk categories based on core damage frequency.
- Minimum compensatory actions as defined by the risk category are provided in WPG-1. Examples of compensatory actions include use of a mini-schedule, continuous work until the risk category is reduced, or Plant Manager approval depending on the risk category.
- WPG-1 also states that consideration of weather conditions should be a part of the decision process utilized by the shift supervisor prior to allowing the removal of equipment from service for planned maintenance.

In the future, the elements of the WPG-1 as presented above, may be modified (e.g. to accommodate improvements in technology), however, the quality of risk management will be maintained.

Probabilistic Safety Assessment Updates:

A major update of the probabilistic risk assessment (i.e., the DBNPS "Probabilistic Safety Assessment") was completed in 1999. The objectives of the update included the following:

- To ensure that the models comprising the Probabilistic Safety Assessment (PSA) accurately reflect the current plant, including its physical configuration, operating procedures, maintenance practices, etc.
- To take into account more recent operating experience with respect to the frequency of plant transients and the failure rates of potentially important components.
- To correct items uncovered subsequent to the Individual Plant Examination (predecessor to the PSA) submittal.
- To address weaknesses identified in past reviews.

This detailed update provided the verification that the current DBNPS PSA reflects the as-built, as-operated plant. Additionally, subsequent to the update a configuration control process has been put in place to maintain the PSA as close as possible to the actual plant configuration. The original calculations for the Topical Report were performed prior to the completion of this update. However, after the update was completed the calculations were formally revised using the updated PSA, and the information provided within the Topical Report remained bounding.

Peer Review Process:

To ensure the overall quality of the Individual Plant Examination (IPE), the DBNPS conducted two different types of peer reviews: independent in-house reviews and external reviews. The in-house reviews were conducted in several different stages. First, each analytical task went through a peer review within the project team. This review also included a review by contract personnel. Subsequent reviews were performed by individuals within the DBNPS who had specific technical expertise in applicable areas. Finally, the project manager and other cognizant DBNPS department managers performed reviews.

Independent, in-house technical reviews of the documents, analysis, and results for all tasks associated with the IPE were performed. These reviews were performed for the development of initiating events, the event trees and corresponding success criteria, all system fault-trees, the various data bases, the human reliability analysis, and the recovery analysis. In addition to reviews performed by the various engineering departments, licensing engineers, training personnel, operations and maintenance organization personnel, and previously licensed senior reactor operators were available to perform reviews, as needed.

During development of the system fault trees, independent review teams were formulated to review the fault trees in detail. Individuals with different types of technical expertise were involved; typically, individuals from Systems Engineering, Maintenance, and Operations were included in the review teams. Other individuals who had performed specialized work on certain systems or in some way could provide valuable input were also included on the review teams, as appropriate. Each team was tasked with ensuring that the model provided an accurate representation of the system design and actual operating and maintenance practices. A previously licensed senior reactor operator, who was familiar with the plant safety analyses, was also responsible for reviewing each individual system fault-tree model.

Development of the reliability data bases was performed by an individual, with review of each element by at least one other individual. The sequence quantification was distributed between and performed by two individuals, each of whom reviewed the work performed by the other. A similar approach was used in reviewing the recovery analysis.

Review of the IPE back-end analysis was handled in a similar fashion. One engineer was responsible for developing the MAAP input deck with subsequent independent reviews of each input parameter by at least one other engineer. Each MAAP analysis was performed by, and separately reviewed by, individuals who had been trained in the use of the MAAP code. Additionally, MAAP runs and the quantification of the containment event tree were reviewed in group work sessions by key IPE team members as well as other individuals in the Nuclear Engineering Safety Analysis Unit.

Comments received as part of the reviews were incorporated as appropriate into the various IPE work packages and final report. As a result of the process outlined above, every individual technical task that comprised the IPE went through at least one independent in-house review, and in most cases there were multiple stages of review.

In addition to the in-house reviews that were performed, the DBNPS contracted with Duke Engineering & Services, Inc., and the Duke Power Company to perform an external peer review. Duke personnel were selected because of their previous experience in the overall performance and application of probabilistic risk assessment methods and their familiarity with plants designed by Babcock & Wilcox. Their review was conducted in a manner that was consistent with a procedure developed for the Electric Power Research Institute. The results of this review were documented in a separate report. The overall results of that review were very positive. Among the high-level comments in the Duke review were the following:

- The technical quality of the completed IPE activities is excellent. Planned activities appear to be well thought out and would be expected to work out with the same quality.
- Documentation is also excellent. The level of documentation makes it easy for someone to reproduce and trace the derivation of the results. At the same time it is being kept to a minimum to eliminate waste caused by excessive documentation.

- The planned report appears to address the NRC submittal guidelines.

All technical comments on the front-end and back-end analyses were evaluated and addressed prior to completion of this License Amendment Request.

In addition to the above reviews, the DBNPS PSA was reviewed by the B&W Owners Group Risk-Based Applications Committee. It was the assessment of the review team that the DBNPS PSA can be effectively used to support applications involving risk significant determinations supported by deterministic analysis.

PSA Quality Assurance Methods:

Documentation of the DBNPS PSA technical elements includes initiating event analysis, sequence development, success criterion, systems analysis, data analysis, human reliability analysis, quantification and results. Information sources, methodologies and assumptions made in the course of the analysis are included in the documentation. The documentation indicates the person who developed the product and the person who reviewed or otherwise verified the appropriateness by an associated signature.

When changes are made to any element of the PSA, a revision is made to the documentation that includes a summary of the changes that were made. The change documentation also indicates the name of the person who performed the revision and the person who independently reviewed the revision by an associated signature.

Results of Reviews with Respect to this License Amendment Request:

DBNPS Calculation C-NSA-99.16-21 contains the documentation of analysis performed for Topical Report BAW-2295A and this License Amendment Request. Attached to this calculation is a list of the cut sets that were reviewed as part of the calculation process. The review verified that the calculation adequately modeled the effects of the Low Pressure Injection and Containment Spray Systems unavailability.

Maintenance Rule:

As discussed earlier, the DBNPS' WPG-1 is consistent with the maintenance rule in 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This includes the scheduling of on line maintenance as within the boundaries established by implementation of the maintenance rule.

Proposed TS Bases Section 3/4.5.2 and 3/4.5.2 Changes

The discussion proposed to be added to TS Bases Section 3/4.5.2 and 3/4.5.3 is for two purposes:

1) To explain the makeup of the ECCS subsystems in TS LCO 3.5.2, and 2) to discuss the flexibility in plant operations afforded by the LCO and Action when ECCS subsystem component(s) are inoperable.

The first item is accomplished by explaining each ECCS subsystem consists of a HPI System train, a LPI System train (including the associated decay heat cooler), and the required flowpaths from the BWST or Containment Emergency Sump. This proposed addition clarifies the meaning of the existing phrase "ECCS subsystems" and does not modify the plants' physical configuration or operation.

The second item incorporates text similar to that contained in the NRC's NUREG-1430, Revision 1, "Standard Technical Specifications for Babcock and Wilcox Plants," Bases B3.5.2 for the ECCS. This text discusses the intent of the LCO Actions to maintain a combination of ECCS equipment to ensure 100% of the safety injection flow equivalent to 100% of a single subsystem of ECCS remains available.

For example, either HPI train combined with either LPI train (including its associated decay heat cooler) could provide the required 100% injection flow. This proposed TS Bases discussion reflects the allowance of the LCO to provide the operation of the plant with flexibility in using redundant trains of equipment as long as the design analyses requirement of 100% injection flow is met. Accordingly, these proposed changes maintain the requirements of the Design Bases Accident analyses and have no adverse effect on nuclear safety.

Non-Emergency Functions of the LPI System

At the DBNPS, the LPI System, including the decay heat coolers, also have non-emergency functions. In the Decay Heat Removal (DHR) mode, it is used to remove reactor core decay heat during shutdown and refueling operations. The DHR System takes suction from the RCS hot leg and returns it back to the RCS after passing it through the requisite DHR coolers. However, this function is not applicable to Modes 1, 2, or 3 and, therefore, is outside the scope of the proposed changes. The requirements of TS 3/4.5.3 – "ECCS Subsystems – $T_{avg} < 280°F$" (Mode 4), 3/4.9.8.1 – "Decay Heat Removal and Coolant Circulation – All Water Levels" (Mode 6), and 3/4.9.8.2, "Low Water Level," (Mode 6) are not affected by the proposed changes and would remain applicable.

The LPI System trains may be realigned for use as a backup system for spent fuel cooling. This non-emergency use of the LPI System as a contingency backup for the Spent Fuel Pool Cooling pumps is not affected by the proposed changes to increase the Allowed Outage Time to seven days. The proposed Allowed Outage Time for the LPI System,

which is an engineered safety features system, has been demonstrated in Topical Report BAW-2295A, Revision 1 to be acceptable. The non-emergency function of the LPI System is bounded by the justification provided by the Topical Report. Accordingly, the proposed changes will not have an adverse effect on nuclear safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, (DBNPS) Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because, as demonstrated in the Babcock and Wilcox Owners Group's Topical Report BAW-2295A, Revisions 1 and 2, *Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems,* no accident initiators, conditions, or assumptions are affected by the proposed changes to extend the allowed outage time (AOT) from 72 hours to 7 days for one inoperable train of Low Pressure Injection (LPI) in Technical Specification (TS) 3/4.5.2 Emergency Core Cooling Systems – ECCS Subsystems – $T_{avg} \geq 280°F$ or Containment Spray in TS 3/4.6.2.1, Containment Systems – Depressurization and Cooling Systems – Containment Spray System. The proposed change to TS Bases Section 3/4.5.2 and 3/4.5.3 are discussions of the present TS Limiting Condition for Operation (LCO) which do not affect the probability of an accident.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because an extension in the allowable outage time from 72 hours to 7 days for one inoperable train will not affect any previously evaluated accidents. The proposed changes to the TS Bases discuss the present TS LCO and do not affect the consequences of an accident. The proposed changes do not alter the source term, containment isolation, or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any previously evaluated because no new failure mode or transient is introduced since the proposed changes do not involve a plant modification or allow operation of any plant systems, structures, or components in a manner not addressed in the DBNPS Design Basis Accident analyses.

3. Not involve a significant reduction in the margin of safety because extending the allowed outage time to 7 days for one inoperable train does not impact any assumptions or inputs in the DBNPS Updated Safety Analysis Report. The proposed changes have been evaluated and determined that the extended allowed outage time is consistent with safe operation considering the redundant systems of required features and the administrative controls in place for removing this equipment from service. The proposed TS Bases changes reflect the existing TS LCO and, therefore, do not reduce a margin of safety.

CONCLUSION:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENT: Attached are the proposed marked-up changes to the Operating License.

REFERENCES:

1. BAW-2295A, Revision 1 and Revision 2, *Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems,* October 1997.

2. DBNPS *Updated Safety Analysis Report* (USAR) through Revision 21.

3. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 243.

4. DBNPS System Description for *Reactor Coolant System*, SD-039A.

5. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific-Changes to the Current Licensing Basis, July 1998.*

6. Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, August 1998.*

7. NUREG-1430, Revision 1, *Standard Technical Specifications – Babcock and Wilcox Plants,* April 1995.

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. ~~With one ECCS subsystem inoperable, restore the inoperable subsystem 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.

NFORMATION ONLY

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

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SURVEILLANCE REQUIREMENTS (Continued)

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.

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SURVEILLANCE REQUIREMENTS (Continued)

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.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the BWST on a containment spray actuation signal and manually transferring suction to the containment emergency sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 7 days ~~72 hours~~ or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system 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.

b. At least once each REFUELING INTERVAL, by:

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

   2. Verifying that each spray pump starts automatically on a SFAS test signal.

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SURVEILLANCE REQUIREMENTS (Continued)

c. Deleted

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES (Continued)

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.

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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.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

The surveillance requirement which verifies that only one door in each air lock can be opened at a time is not part of the Containment Leakage Rate Testing Program. Therefore, its test frequency is subject to the provisions of Specification 4.0.2.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psi and 2) the containment peak pressure does not exceed the design pressure of 40 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 37 psig. The limit of 1 psig for initial positive containment pressure will limit the total pressure to 38 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

Deleted

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

Maintaining the containment purge supply and exhaust isolation valves closed with control power removed at all times during MODES 1, 2, 3 and 4 provides assurance that the safety function of containment isolation is maintained in the event of a LOCA.

The ACTION statement assures that at least one containment purge supply and exhaust isolation valve is closed in each containment penetration and provides reasonable time to permit closure of an open valve.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment

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leakage rate are consistent with the assumptions used in the safety analyses.

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.

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 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 valves 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.

3/4.6.2.2 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the required time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Containment isolation valves and their required isolation times are addressed in the USAR. The opening of a closed inoperable containment isolation valve on an intermittent basis during plant operation is permitted under administrative control. Operating procedures identify those valves which may be opened under administrative control as well as the safety precautions which must be taken when opening valves under such controls.

The containment purge supply and exhaust system isolation valves are considered OPERABLE with respect to containment isolation when they meet the requirements of Specification 3.6.1.7.

Docket Number 50-346
License Number NPF-3
Serial Number 2667
Attachment 2

TYPED TECHNICAL SPECIFICATION

REPLACEMENT PAGES

(4 Pages Follow)

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.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the BWST on a containment spray actuation signal and manually transferring suction to the containment emergency sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system 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.

b. At least once each REFUELING INTERVAL, by:

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

   2. Verifying that each spray pump starts automatically on a SFAS test signal.

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.

Docket Number 50-346
License Number NPF-3
Serial Number 2667
Enclosure 2

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
Ensure the four compensatory measures, as listed on page 7 of the Safety Assessment and Significant Hazards Consideration, are implemented to lower the risk impact of Low Pressure Injection (LPI) and Containment Spray System outages with the plant on line.	Prior to first use of the 7 day Allowed Outage Time for LPI TS 3/4.5.2 and Containment Spray System TS 3/4.6.2.1.