1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2772 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s). Prior to entering MODE 1 for Cycle 5, auxiliary feedwater system OPERABILITY shall be determined without consideration of the status of the startup feedwater system.

Emergency Request
Circumstances and Justification

The following explains the circumstances leading to this emergency request and justification of the need for prompt NRC action.

Potential Consequences of Auxiliary Boiler Use During Heatup and Zero Power Testing

The Davis-Besse plant has been shutdown for 95 days during which time approximately one-third of the reactor core has been replaced with new fuel. After this extended shutdown period and corresponding reduction in decay heat, insufficient thermal energy remains in the RCS to maintain reliable operation of the Main Feedwater Pumps for plant startup. Under these low decay heat conditions (coupled with the unavailability of the SUFP), it is necessary to use the Auxiliary Boiler to provide steam to the Main Feedwater Pumps.

When the Auxiliary Boiler is supplying steam to the Main Feedwater Pumps, a boiler trip will result in loss of feedwater which actuates the safety grade Steam and Feedwater Rupture Control System (SFRCS). The SFRCS will then cause a reactor trip (reportable to the NRC) and actuation of the AFW system. This will change the steam generator level which will affect the RCS pressure and temperature. Such a change in RCS conditions will have a major impact on the physics testing program because the time required to re-established test conditions could be between eight and 24 hours for each such trip.

The Auxiliary Boiler is normally used to supply other auxiliary system steam loads when main steam is not available Past experience has shown that use of the boiler when supplying steam to the Main Feedwater Pumps in addition to other auxiliary steam requirements can result in sufficiently high steam demand to affect the performance of the boiler.

Thus, it can be determined that use of the Auxiliary Boiler can result in:

1. Increased challenges to safety systems (SFRCS, AFW).

2. Increased reported reactor trips.

3. Significant lengthening of the zero power physics testing program, with a corresponding increase in need for replacement power.

Based upon the above, it is clearly undesirable to use the Main Feedwater Pumps and Auxiliary Boiler to supply feedwater in lieu of the Startup Feedwater Pump.

Chronology of Events Necessitating An Emergency Request

In May 1984, while performing a review related to the SUFP system it was discovered that use of the SUFP presented a high/moderate energy line

break hazard to the Auxiliary Feedwater (AFW) system. TEDs internal mechanism (Surveillance Report) for capturing and tracking such potential problem items through resolution was implemented. TED committed to the NRC to implement certain operational restrictions and compensatory measures concerning operation of the SUFP.

Efforts were then initiated to resolve the problem from a technical standpoint. A lead engineer was assigned to follow this problem and Bechtel (TEDs architect/engineer) was requested to develop potential options for early resolution of this problem. This engineering review effort resulted in a determination that no viable solution could be developed and implemented in the time remaining before completion of the Fall refueling outage.

Concurrent with the above engineering effort, TED personnel performed further safety reviews to quantity the significance of this problem to the Davis-Besse design basis. A second engineer was assigned to perform an analysis to quantify the significance of the risk to the AFW system due to the use of the SUFP.

This review of risk resulted in a determination that the real risk to the AFWS was not significant and that continued use of the SUFP for normal startups and shutdowns was technically justified. This determination was documented in a safety evaluation and was reviewed and approved by TEDs onsite and corporate safety review boards.

Work then commenced on a request for permission from the NRC to use the SUFP for startup and shutdowns. TED recognized the operational importance of the SUFP to Davis-Besse and took great effort to assure, through an extensive technical review and comment period, that the technical review was complete and accurate. The request for a Safety Evaluation Report was submitted on October 18, 1984. At the time of submittal, TED considered its timing to be adequate to allow appropriate NRC review and subsequent approval prior to the restart of Davis-Besse.

It was subsequently determined, however, that a license amendment was the appropriate licensing vehicle to resolve this problem. Accordingly, TED expedited its internal review and submittal of a license amendment request.

In addition to the specific activities mentioned above, discussions were held with both NRR and OI&E personnel to keep them appraised of the situation. This chronology demonstrates that Toledo Edison (TED) has shown prudent and proper management prioritization with regards to efforts to achieve a timely resolution of the SUFP issue. Since sufficient time does not remain prior to the present scheduled need for the SUFP to meet the noticing procedure of 10 CFR Part 50, Section 50.91 cannot be met and an emergency circumstance exists. Toledo Edison firmly believes that an expedited review and approval of this license amendment is warranted.

Justification for an Emergency Request

Toledo Edison's share of the Davis-Besse station is 428 MWe which is approximately 32 percent of the projected winter season demand of 1350 MWe. An extended unavailability of the Davis-Besse station will create the need for TED to purchase approximately 150 MWe until the Davis-Besse station is synchronized to the grid. In addition, Cleveland Electric and Illuminating Company will be required to purchase 170 MWe replacement power. These figures are contingent upon winter weather conditions within the combined service areas and the continued operation of other electric generating units. The cost of replacement power will most likely be high if other electric utilities are experiencing heavy load demand, and the possibility exists that power might not be available should other utilities experience generation problems or should weather conditions cause an increase in electrical demand.

At this time there are no major restraints to the station's scheduled December 28, 1984, synchronization to the grid other than the standard NRC restart approval and NRC approval of this Technical Specification change. Emergency action on the attached amendment request is required to remove the latter restraint.

Based upon the above information, we believe Toledo Edison has acted prudently and expeditiously in its attempts to make a timely application for this amendment and resolve the SUFP issue. Furthermore, in recognition of the power demand situation in our service area and the need for power from the Davis-Besse station, we firmly believe an emergency request and NRC approval is justified.

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Serial No. 1105
Attachment C

Startup Feedwater Pump Issue Chronology

1984

May	June	July	August	September	October	November
Problem Identified.	Problem Studied. LER Generated. Solutions Identified.		SRB/CNRB Review	Meeting with NRC.	Submittal to NRC. NRC Review.	NRC Review. License Amendment Request.

Date	ACTIVITY DESCRIPTION
May 15, 1984	Toledo Edison prepares Surveillance Report on Startup Feedwater Pump (SUFP) operating contrary to USAR description.
May-June, 1984	Toledo Edison review of SUFP situation and preparation of a safety evaluation.
June 18, 1984	Toledo Edison generates Deviation Report on SUFP situation.
July 11, 1984	Toledo Edison initiates implementation of Facility Change Request (FCR) 84-119 to perform a 10 CFR 50.59 review and obtain NRC concurrence for interim use of the SUFP with administrative procedures.
July 18, 1984	Toledo Edison Licensee Event Report (LER) No. 84-009 submitted reporting the discovery of the SUFP concern.
August 3, 1984	Probabilistic Risk Assessment (PRA) - based justification for SUFP operation completed.
August 8, 1984	Station Review Board (SRB) meeting and approval of FCR 84-119.
August 10, 1984	Company Nuclear Review Board (CNRB) meeting and approval of FCR 84-119.
Sept. 17, 1984	Toledo Edison internal meeting on SUFP modification.
Sept. 19, 1984	Toledo Edison meeting with NRC Staff regarding SUFP problem.
Sept. 26, 1984	Toledo Edison internal Design Review meeting with Operations Department.

Date	ACTIVITY DESCRIPTION
October 18, 1984	Toledo Edison letter (Serial 1070) submitted to the NRC, with the PRA and Safety Evaluation, requesting expedited review of the interim use of the SUFP.
November 5, 1984	NRC Staff notifies Toledo Edison that a License Condition on the SUFP is necessary.
November 7, 1984	Toledo Edison initiates implementation of FCR 84-192, on the proposed SUFP License Condition (2.C.(3)(t)).
November 7, 1984	SRB meeting and approval of FCR 84-192.
November 9, 1984	CNRB meeting and approval of FCR 84-192.
November 12, 1984	Toledo Edison letter (Serial 1100) submitted to the NRC requesting a License Condition for the SUFP.
November 20, 1984	NRC Staff notifies Toledo Edison that a 30-day public notice period will be necessary for the proposed License Condition.
Nov. 20-29, 1984	Toledo Edison/NRC discussions concerning a proposed license amendment.
November 21, 1984	Toledo Edison letter (Serial No. 1093) submitted general system description and flow diagram for the proposed new SUFP.
November 28, 1984	Toledo Edison initiates implementation of FCR 84-208, on an emergency request for a license amendment modifying Technical Specification Section 1.6.
November 28, 1984	SRB meeting and approval of FCR 84-208.
November 29, 1984	CNRB meeting and approval of FCR 84-208.
December 3, 1984	Toledo Edison requests NRC approval of a change to Technical Specification Section 1.6 on an emergency basis.
December 19, 1984	Scheduled Davis-Besse need for SUFP for zero power physics testing.

Event Evaluation

A description of the postulated event used to evaluate the interim (December 19, 1984 until entering Mode 1) Startup Feedwater Pump (SUFP) use is as follows:

A. Initial Plant Conditions:

1. Reactor Coolant System (RCS) Temperature - The startup feedwater pump will be used to support the RCS heat-up from a RCS temperature of 280°F (minimum temperature for entry into Mode 3) to 530°F.

2. RCS Pressure - Approximately 2155 psi.

3. Relevent Equipment Status - Steam and Feedwater Rupture Control System (SFRCS) and Auxiliary Feedwater (AFW) system lined up and available on demand; Emergency Core Cooling System (ECCS) subsystems available on demand; SUFP lined up and in operation; Reactor Coolant Pumps (RCP) operating.

4. Calculated decay heat load in the core - 2.3 MWt (Equivalent to effective heat available from the 113 pre-Cycle 5 fuel assemblies after 95 days shutdown.)

B. Evaluation of Postulated Failure Events:

1. The assumed initiating event being evaluated is a loss of all feedwater (main, auxiliary and startup) with no loss of offsite power. For this case, offsite power is assumed to remain available for maximizing the amount of energy contained in the RCS. The non-seismic suction and discharge piping of the SUFP is assumed to fail in such a manner as to render both trains of Auxiliary Feedwater (AFW) inoperable. This is postulated to result in a total loss of secondary side cooling. The interruption of water to the Steam Generators will result in a reactor trip from the Anticipatory Reactor Trip System (ARTS) due to a Steam and Feedwater Rupture Control System (SFRCS) Actuation signal. The SFRCS trip will be generated as a result of either Low-Steam Generator Level or Steam to Feedwater Differential Pressure. Due to the reactor trip, the Control Room operators will implement the appropriate steps of Station Emergency Procedure EP 1202.01, "RPS, SFAS, SFRCS Trip or SG Tube Rupture".

   Since it is assumed that AFW is unavailable due to the SUFP piping failure, RCS pressure will begin to rise due to the lack of heat removal. As RCS pressure approaches the sub-cooling limits the operator will trip the Reactor Coolant Pumps as per EP 1202.01. This will limit the energy input to the RCS to decay heat only.

EP 1202.01 directs the operator to respond to a loss of main and auxiliary feedwater by taking the following actions:

1. Open the Pilot Operated Relief Valve.
2. Open the RCS Hot Leg High Point Vents
3. Open the Pressurizer Vent
4. Start the Makeup Pumps
5. Ensure that the High Pressure Injection Pumps start when the RCS pressure drops to 1650 psig.

For the purpose of conservative evaluation, acceptability of handling this postulated event is ensured assuming only safety grade equipment with the imposition of a limiting single failure. For this postulated event the No. 2 Diesel Generator is assumed to fail and, therefore, removes one High Pressure Injection (HPI) Pump and the pressurizer vent path. This vent path could remove more energy than the Hot Leg High Point Vents due to its steam space location.

Venting the RCS to depressurize, and injecting coolant into the RCS would continue within appropriate pressure temperature limits until the Decay Heat Removal System can be put in service. EP 1202.01 identifies several combinations to accomplish this process. For this evaluation the minimum safety grade equipment (Table 2) capability available with the single failure of No. 2 Diesel Generator would be one HPI train and two Hot Leg High Point Vents (Table 1, Option 1). This combination of one HPI pump and two Hot Leg High Point Vents results in a 4.8 MWt removal, more than twice the required heat removal capability. No excess inventory loss would result; therefore, no core uncovery would take place. Therefore, even in the unlikely event of a loss of startup feedwater concurrent with a loss of both trains of Auxiliary Feedwater, the concern can safely be mitigated.

Table 1: Comparison of Selective Options Available*

Option	Description	Estimated Flow Through RCS (gpm)	Initial Rate of Heat Removal (MWt)	Required Heat Removal (MWt)	Excess Heat Removal Available (MWt)
1	1 HPI Train with 2 RCS Hot Leg High Point Vents	78	4.8	2.3	2.5
2	1 Makeup Train with 2 RCS Hot Leg High Point Vents	90	5.5	2.3	3.2
3	1 HPI train with 1 RCS Hot Leg High Point Vent	39	2.4	2.3	0.1

* Options utilizing two pumps will result in more conservative flows and heat removal rates.

Table 2: Potential Cooling Components

Component	Seismic	1E Powered	Onsite Non-1E Powered	Power Supply Description	Comments
PORV	No	-	Yes	No. 1 Train Non-Essential 120 VAC	
PORV Block Valve	No	Yes	-	No. 1 Train Essential 480 VAC	
Pressurizer Vent RC 200	Yes	Yes	-	No. 2 Train Essential 480 VAC	
Pressurizer Vent RC 230A	Yes	Yes	-	No. 2 Train Essential 480 VAC	
Hot Leg No. 1 High Point Vent	Yes	Yes	-	No. 1 Train Essential 120 VAC, 120 VDC	Each High Point Vent has an AC powered valve and a DC powered valve in series.
Hot Leg No. 2 High Point Vent	Yes	Yes	-	No. 2 Train Essential 120 VAC, 120 VDC	Each High Point Vent has an AC powered valve and a DC powered valve in series.
Makeup Pump No. 1	No	Yes	-	No. 1 Train Essential 4160 VAC	
Makeup Pump No. 2	No	Yes	-	No. 2 Train Essential 4160 AC	
HPI Pump No. 1	Yes	Yes	-	No. 1 Train Essential 4160 VAC	
HPI Pump No. 2	Yes	Yes	-	No. 2 Train Essential 4160 VAC	
Letdown Coolers	No	NA	NA		Cooled by fully safety grade, seismic, component cooling water. Letdown is isolated by the Safety Features Actuation System on low RCS Pressure (1650 psig).

Safety Evaluation

The requested change to Technical Specification Section 1.6 would allow the Davis-Besse Nuclear Power Station Unit No. 1 to complete reactor coolant heatup and conduct zero power physics testing prior to entering Mode 1 of the new fuel cycle (Cycle 5) utilizing the present configuration of the SUFP and associated pipelines and valves.

The safety function of the AFWS is to provide an alternate source of water to the steam generators for heat removal. The SUFP system itself performs no safety function. It is, however, used as a backup to the main and auxiliary feedwater systems for supplying water to the steam generators in case of the total loss of these two systems.

The SUFP is located in Room 238 which is the same room as one of the auxiliary feedwater pumps (AFP), pump 1-2. During the review of the high/moderate energy line break criteria as it relates to the AFW rooms, it was determined that the SUFP system can jeopardize AFP 1-2 in the event of a pipe leak or rupture.

The SUFP system is non-seismic downstream of valve FW91. The suction of the SUFP system utilizes the Deaerator or the Condensate Storage Tank (CST) as its water source. The normal source is the Deaerator. It also uses the non-seismic Turbine Plant Cooling Water (TPCW) system for pump cooling. The line from the Deaerator to the SUFP also runs through Room 237. Room 237 contains AFP 1-1. The discharge line from the SUFP is aligned to the inlet of the high pressure feedwater heaters. In the past, these systems were not valved off so as to provide immediate backup for the Main Feedwater System (MFWS) if needed.

Since this problem was identified, the SUFP system and the TPCW system in Rooms 237 & 238 have been isolated. This isolation has been accomplished by closing valve FW91 from the CST, valve FW32 from the Deaerator, valves CW196 and CW197 used for pump cooling, and valve FW106 in the discharge line.

The concern with the location of the SUFP is the potential for pipe whip and jet impingement in Room 238 and flooding and high temperature in Rooms 237 and 238. The concern with the TPCW system is the potential for flooding these rooms. These concerns will only be realized when the SUFP system is available for operation.

During the period that the startup pump is operating, the suction piping to the SUFP is a moderate energy line based on the criteria in USAR Section 3.6 which states that a line outside containment operating above 275 psig or 200°F is a moderate energy line. The discharge piping from the SUFP is a high energy line based on the USAR Section 3.6 which states that a line outside containment operating both above 275 psig and 200°F is a high energy line. If a high energy line is in service more than six hours, Section 3.6 requires that it must be analyzed for pipe rupture. The SUFP system could be in operation for as long as 336 hours (14 days) during reactor coolant heatup and zero power physics testing. The TPCW system supply and return piping are neither moderate nor high energy lines but, since the lines are non-seismic, a flooding concern remains.

It has been postulated that during a seismic event the pipes would rupture in such a manner as to damage AFP 1-2 or possibly AFP 1-1 due to flooding. It has also been postulated that with a high energy pipe break the sheared pipe could damage AFP 1-2 due to jet impingement or pipe whip and would cause high temperatures and pressure in Room 238. A moderate energy pipe break could damage either AFP 1-1 or 1-2 due to flooding and high temperature in either Room 237 or 238. A rupture/break in the TPCW piping could damage either AFP 1-1 or 1-2 due to flooding in either Room 237 or 238.

A Probability Risk Assessment (PRA) study (Attachment G) has been performed since this situation was discovered. The PRA documents that the worst case probability for a rupture/break in the SUFP and the TPCW piping causing the failure of AFP 1-2 is of the order of 1.45E-6 per the 14-day period (conservatively the period of time under this amendment during which the SUFP would be operating). The probability for failure of AFP 1-1 in Room 237 is smaller due to less SUFP piping in the room. The probability for failure of the AFWS due to pipe rupture/break in the SUFP and the TPCW system is documented in Calculation Number C-NSA-45.02-03. This probability is insignificant in light of the AFWS unavailability on the order of 1E-2/yr. for each train which was submitted to the NRC in December, 1981.

Although these risks to the AFWS from the SUFP systems are considered insignificant, the SUFP suction and discharge piping were hydrotested in the Fall, 1984, to the original acceptance criteria (ANSI B31.1) to ensure the integrity of the SUFP suction and discharge piping. In addition, certain precautionary measures will be observed during SUFP operation, when the SUFP is not being used as a source of auxiliary feedwater. An operator shall be positioned at the AFW room area when the SUFP is operating in Modes 2 and 3. Upon indication of a pipe leak the operator will either trip the SUFP locally or contact the control room to trip the SUFP. This may not reduce the probability for a pipe break in the SUFP system, however, it will reduce significantly the impact of a SUFP system failure resulting in a AFWS failure. He would then close all SUFP isolation valves which are external to the AFW rooms. This operator action is being taken since piping leaks are expected to occur prior to any complete piping rupture. If the SUFP is being used as a source of auxiliary feedwater, specific direction appropriate to the situation will be provided to the operator by the shift supervisor.

In the unlikely event that the SUFP system should fail and render the AFWS inoperable, Calculation Number C-NSA-45.02-005 shows that the reactor decay heat (2.3 MWe) after a 95 day shutdown could be adequately removed (see Attachment D). A heat load of 2.3 MWt requires a flow of approximately 40 gpm for decay heat removal. Three alternatives for the required decay heat removal have been investigated and are presented in Table 1 (see Attachment D). The first, a combination of one HPI and two hot leg high point vents (this combination is both safety grade and meets the single failure criteria) is capable of a flow of 78 gpm and a heat removal of 4.8 MWt. This is a factor of 1.1 times more than is necessary. The second, a combination of one makeup pump and two hot leg high point vents is capable

of a flow of 90 gpm and a heat removal rate of 5.5 MWt. This is a factor of 1.4 times more than is necessary for decay heat removal. The third, a combination of one HPI with one hot leg high point vent, is capable of a flow of 39 gpm and a heat removal rete of 2.4 MWt. This is also more than is necessary for decay heat removal.

The events postulated above result in a plant response that assures the ability of residual heat removal utilizing safety grade equipment under single failure conditions. This is done without violating full design limits on RCS pressure boundary design conditions and therefore meets the General Design Criteria 34. Additionally, in 1979, an analysis was performed by Babcock & Wilcox relating to loss of all feedwater events. An NRC evaluation of the B&W analysis was provided showing adequate residual heat removal capability with the injection of coolant into the RCS to assure no core uncovery. This evaluation recognized that the initiating event was highly unlikely but that never-the-less the results were acceptable. Based on the above, it is concluded that this Technical Specification change does not involve an unreviewed safety question.

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Attachment F

Significant Hazards Consideration

The proposed license amendment would allow the Davis-Besse Nuclear Power Station Unit No. 1 to conduct reactor coolant heatup and zero power physics testing prior to entering Mode 1 of the new fuel cycle (Cycle 5) utilizing the present configuration of the Startup Feedwater Pump (SUFP) and its associated pipelines and valves. As explained below, this proposed amendment does not involve a significant hazard.

The safety function of the AFWS is to provide an alternate source of water to the steam generators for heat removal. The SUFP system itself performs no safety function. It is, however, used as a backup to the main and auxiliary feedwater systems for supplying water to the steam generators in case of the total loss of these two systems.

The SUFP is located in Room 238 which is the same room as one of the auxiliary feedwater pumps (AFP), pump 1-2. During the TED review of the high/moderate energy line break criteria as it relates to the AFW rooms, it was determined that the SUFP system can jeopardize AFP pump 1-2 in the event of a pipe leak or rupture.

The SUFP system is non-seismic downstream of valve FW91. The suction of the SUFP system utilizes the Deaerator or the Condensate Storage Tank (CST) as its water source. The normal source is the Deaerator. It also uses the non-seismic Turbine Plant Cooling Water (TPCW) system for pump cooling. The line from the Deaerator to the SUFP also runs through Room 237. Room 237 contains AFP 1-1. The discharge line from the SUFP is aligned to the inlet of the high pressure feedwater heaters. In the past, these systems were not valved off so as to provide immediate backup for the Main Feedwater System (MFWS) if needed.

Since this problem was identified, the SUFP system and the TPCW system in Rooms 237 & 238 have been isolated by TED. This isolation has been accomplished by closing valve FW91 from the CST, valve FW32 from the Deaerator, valves CW196 and CW197 used for pump cooling, and valve FW106 in the discharge line.

The concern with the location of the SUFP is the potential for pipe whip and jet impingement in Room 238 and flooding and high temperature in Room 237 or 238. The concern with the TPCW system is the potential for flooding these rooms. These concerns will only be realized when the SUFP system is available.

During the period that the startup pump is operating, the suction piping to the SUFP is a moderate energy line based on the criteria in USAR Section 3.6 which states that a line outside containment operating above 275 psig or 200°F is a moderate energy line. The discharge piping from the SUFP is a high energy line based on the USAR Section 3.6 which states that a line outside containment operating both above 275 psig and 200°F is a high energy line. If a high energy line is in service more than six hours,

Section 3.6 requires that it must be analyzed for pipe rupture. The SUFP system could be in operation for as long as 336 hours (14 days) during reactor coolant heatup and zero power testing. The TPCW system supply and return piping are neither moderate nor high energy lines but, since the lines are non-seismic, a flooding concern remains.

It has been postulated that during a seismic event the pipes would rupture in such a manner as to damage AFP 1-2 or possibly AFP 1-1 due to flooding. It has also been postulated that with a high energy pipe break the sheared pipe could damage AFP 1-2 due to jet impingement or pipe whip and would cause high temperatures and pressure in Room 238. A moderate energy pipe break could damage either AFP 1-1 or 1-2 due to flooding and high temperature in either Room 237 or 238. A rupture/break in the TPCW piping could damage either AFP 1-1 or 1-2 due to flooding in either Room 237 or 238.

No Significant Increase in Probability or Consequences

A Probability Risk Assessment (PRA) study has been performed since this situation was discovered. The PRA documents that the worst case probability for a rupture/break in the SUFP and the TPCW piping causing the failure of AFP 1-2 is of the order of 1.45E-6 per the 14-day period (conservatively the period of time under this amendment during which the SUFP would be operating). The probability for failure of AFP 1-1 in Room 237 is smaller due to less SUFP piping in the room. The probability for failure of the AFWS due to pipe rupture/break in the SUFP and the TPCW system is documented in Calculation Number C-NSA-45.02-03. This probability is insignificant in light of the AFWS unavailability on the order of 1E-2/yr. for each train which was submitted to the NRC in December, 1981. Accordingly, this amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated (loss of feedwater).

Although risks to the AFWS from the SUFP system are considered insignificant, the SUFP suction and discharge piping were hydrotested in the Fall, 1984, to the original acceptance criteria (ANSI B31.1) to ensure the integrity of the SUFP suction and discharge piping. In addition, certain precautionary measures will be observed during SUFP operation, when the SUFP is not being used as a source of auxiliary feedwater. An operator shall be positioned in the AFP room area when the SUFP is operating in Modes 2 and 3. Upon indication of a pipe leak the operator will either trip the SUFP locally or contact the control room to trip the SUFP. This may not reduce the probability for a pipe break in the SUFP system, however, it will reduce significantly the impact of a SUFP system failure causing a AFWS failure. He would then close all SUFP isolation valves which are external to the AFP rooms. This operator action is being taken since piping leaks are expected to occur prior to any complete piping rupture. If the SUFP is being used as a source of auxiliary feedwater, specific direction appropriate to the situation will be provided to the operator by the shift supervisor.

No Significant Reduction in Margin of Safety

In the unlikely event that the SUFP system should fail and render the AFWS inoperable, Calculation Number C-NSA-45.02-005 shows that the reactor decay heat (2.3 MWt) after a 95 day shutdown could be adequately removed (see Attachment D). A heat load of 2.3 MWt requires a flow of approximately 40 gpm for decay heat removal. Three alternatives for the required decay heat removal have been investigated. The first, a combination of one HPI pump and two hot leg high point vents (this combination is safety grade and meets the single failure criteria) is capable of a flow of 78 gpm and a heat removal rate of 4.8 MWt. This is a factor of 1.1 times more than is necessary. The second, a combination of one makeup and two hot leg high point vents is capable of a flow of 90 gpm and a heat removal rate of 5.5 MWt. This is a factor of 1.4 times more than is necessary for decay heat removal. The third, a combination of one HPI pump with one hot leg high point vent, is capable of a flow of 39 gpm and a heat removal rate of 2.4 MWt. This is also more than is necessary for decay heat removal. Accordingly, there is not a significant reduction in the margin of safety.

No Creation of the Possibility of a New or Different Accident

An analysis of a total loss of Main and Auxiliary Feedwater was performed, at the NRC's request, by B&W in 1979. This analysis, which showed that a loss of all feedwater did not result in core uncovery and resulted in a calculated peak fuel cladding temperature of less than 800°F, was evaluated by the NRC and the consequences deemed acceptable. This evaluation is documented in Harold Denton's letter to John J. Mattimoe (Sacramento Municipal Utility District) dated June 27, 1979. This previous NRC evaluation of the B&W analysis is applicable to the current Davis-Besse situation. Therefore, the license amendment being requested does not create the possibility of new or different accident from any previously evaluated.

Based on the above information and the attached Safety Evaluation this Amendment Request would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not present a significant hazard.

PRA Based Justification for Operation of the SUFP System

During SUFP system operation, a portion of the SUFP discharge line and the minimum recirculation line renders itself to consideration as a high energy line. Similarly, a portion of the suction piping because of use with the Deaerator Storage Tank water requires consideration as a moderate energy line. The immediate safety impact of lack of such high/moderate energy line considerations is on the Auxiliary Feedwater System (AFWS) since the SUFP is housed in the same room as Auxiliary Feedwater Pump (AFP) 1-2. A portion of the discharge and minimum recirculation lines is routed in this same room. In addition, the suction path from the Deaerator runs through both AFP rooms. With the SUFP in operation while taking suction from the Deaerator the following lengths of pipes may pose a challenge to the AFWS in view of the high/moderate energy line breaks. Furthermore, the non-seismic TPCW piping, in the AFP rooms (listed below), poses a potential flooding concern.

17 feet of 4" discharge line in AFP Room 1-2
18 feet of 1½" minimum recirculation line in AFP Room 1-2
27 feet of 6" suction line in AFW Room 1-2
27 feet of 6" suction line in AFP Room 1-1
62 feet of 4" TPCW line in AFP Room 1-1
40 feet of 4" in AFP Room 1-2
24 feet of less than or equal to 2" in AFP Room 1-2

The overall figure of merit for any one train of the Davis-Besse AFWS is of the order of 1E-2 per year as deduced from the AFWS PRA study (EDS Report No. 02-1040-0195, Revision 1) submitted to the NRC in December 1981. This implies that one train of the Davis-Besse AFWS will be unavailable with a frequency of 1E-2 per year for all initiating events which may require availability of AFWS.

The SUFP could be in operation for as long as 336 hours (14 days) during heatup and zero power physics testing. Assuming the duration of SUFP operation to be 336 hours, the total worst case probability of any break (whether high energy, moderate energy or seismic) in the unanalyzed piping which may challenge the availability of an AFW train is of the order of 1.45 E-6 per the 14-day period. The worst case probability is for AFW train 1-2 because of significantly larger overall length of piping in this room. This evaluation conservatively assumes that any rupture of this non-seismic piping will flood the room to the extent of causing train inoperability with a probability of unity. For further conservatism this assumes that both AFW trains are rendered inoperable.

Since the SUFP system failure as postulated above poses a challenge to the AFW train at a frequency of 1.45 E-6 per the 14-day period, the probability of such SUFP system ruptures/breaks leading to inoperability of an AFW train is insignificant as compared to other failures that may render the AFW system inoperable. It is, therefore, concluded that the above issue poses a very minimal risk to the accomplishment of the safety function of

the AFWS and an extremely negligible risk to public health and safety. Operation of the SUFP system during the time period evaluated above is, therefore, adequately justified.