Point Beach Nuclear Plant

10 CFR 50.59 EVALUATION (NEW RULE)

EVAL 2002-005

Verify EVAL number on all pages

Title of Proposed Activity: Permanent Procedure Changes to Address Potential for Simultaneous Failure of All AFW Pumps

Associated Reference(s) #: CR 01-2278, CR 01-3595 and LER 266/2001-005-00
Calculation WE-0005-06
OM 3.7, Rev. 10, "AOP and EOP Procedures Sets Use and Adherence"
Flowserve Corporation Pump Division letter dated March 2, 2001
Westinghouse Application Data AD 33-760, dated November 1967
EOP-0, Reactor Trip or Safety Injection (Unit 1 and Unit 2), EOP-0.1, Reactor Trip Response (Unit 1 and Unit 2), ECA-0.0, Loss of All AC Power (Unit 1 and Unit 2), and Alarm Response Procedure, ARP C01 A 1-9, Instrument Air Header Pressure Low (Unit 0)

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PART I – DESCRIBE THE PROPOSED ACTIVITY AND SEARCH THE CURRENT LICENSING BASIS

I.1. Describe the proposed facility change, procedure change, test, or experiment which this evaluation is addressing. Parts of an activity for which 10 CFR 50.59 is not applicable or have been pre-screened or screened are to be documented on Forms PBF-1515a, PBF-1515b or PBF-1515c, or on forms associated with the change activity (i.e., PBF-0026a, PBF-0026c, etc.). 10 CFR 72.48 evaluations are documented on Form PBF-1515e. The applicable description from the screening may be provided.

The proposed activity is to make permanent procedure changes that were implemented in response to a condition that was identified where, with a procedure-directed operator action to control steam generator level (which could be accomplished by reducing flow through one or more AFW pumps), concurrent with a loss of instrument air (which would cause the AFW pumps' mini-recirculation valves to fail close), the potential existed for a simultaneous failure of the multi-stage high pressure AFW pumps due to very low or no flow through running AFW pumps. The procedures were initially revised as a compensatory measure to support AFW pump operability. This 50.59 evaluation will review the procedure changes as a permanent change to the procedures as described in the FSAR to confirm consistency with the licensing basis. The permanent procedure changes will restore the AFW pumps to fully operable status.

This evaluation also forms the basis for changes to the FSAR and Technical Specification Bases to clarify that the mini-recirculation valves require instrument air to function and that either a pump minimum flow is maintained or pumps are secured if the valve fails or instrument air is lost.

This evaluation does not credit backup air sources which have been added to all minimum flow recirculation valves. Instrument air accumulator tanks have been installed by MR 02-001 for the 1/2P-29 valves (1/2AF-4002), and the existing nitrogen backup system for the motor-driven auxiliary feed pump discharge valves has been tied in by MR 01-144 for the P-38A/B valves (AF-4007, AF-4014). These backup air sources are not credited ~~because the changes were not designated as safety-related.~~

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Abnormal Operating Procedure AOP-5B, Loss of Instrument Air, includes a step to check that the AFW Pump Mini-recirc Valves are gagged open. However, this step is located in the latter part of the procedure and would not be reached in time to prevent pump failure. Therefore, no credit is taken for this procedure in this evaluation.

FSAR 7.3.3.4, "Manual AFW Flow Control During Plant Shutdown" states: *"The successful operation of the engineered safety features only involves actuation, with one exception. This exception is manually controlling steam generator water level using the auxiliary feedwater pumps during plant shutdown, to remove reactor decay and sensible heat. This manual control involves positioning the auxiliary feedwater flow control valves in order to maintain proper steam generator water level."* The following statements appear in the Point Beach Final Facility Description and Safety Analysis Report (FFDSAR, page 7.5-9 dated May 12, 1969) which was part of the basis for the plant's original operating license: *"The successful operation of the engineered safety features involves only actuation, with one exception. This exception is the steam generator level control function associated with plant cooldown using the auxiliary feedwater pumps. This level control system involves remote manual positioning of feedwater flow control valves in order to maintain proper steam generator water level."*

CR 01-2278, CR 01-3595 and LER 266/2001-005-00 identified an issue that could cause a common mode failure of all auxiliary feedwater pumps. If an accident or event has occurred that is due to or has led to the loss of instrument air, then the auxiliary feedwater pump minimum recirculation control valves 1/2AF-4002 for 1/2P-29, AF-4007 for P-38A, and AF-4014 for P-38B will all fail closed. During this event, it will become necessary for operators to throttle back auxiliary feedwater flow to control steam generator level, especially if all auxiliary feedwater pumps auto start as designed. If it is not recognized that the AFW minimum recirculation valves have closed, and pump discharge valves are closed to control steam generator level (with no minimum flow path), then the pumps would dead head and fail in a short time due to overheating.

The auxiliary feedwater minimum flow recirculation valves do not have a safety function in the open position, however this function is a design function described in FSAR Section 10.2 and Technical Specification Bases B3.7.5. The basis for the recirculation valves not having a safety function in the open position is that all of the auxiliary feedwater pumps will have open discharge valves upon auto-start (even with a loss of instrument air), and thus the minimum flow recirculation line is not needed early in the accident or event. These minimum recirculation valves have an augmented quality function to open (by manual override) to ensure adequate flow though the associated auxiliary feedwater pump during an Appendix R fire within 45 minutes (per Calculation WE-0005-06).

The following procedures were changed: EOP-0, Reactor Trip or Safety Injection (Unit 1 and Unit 2), EOP-0.1, Reactor Trip Response (Unit 1 and Unit 2), ECA-0.0, Loss of All AC Power (Unit 1 and Unit 2), and Alarm Response Procedure, ARP C01 A 1-9, Instrument Air Header Pressure Low (Unit 0). The following procedure changes were made:

1. The step below was added to the Foldout Pages in each of the following procedures: EOP-0, Reactor Trip or Safety Injection (Unit 1 and Unit 2), EOP-0.1, Reactor Trip Response (Unit 1 and Unit 2), ECA-0.0, Loss of All AC Power (Unit 1 and Unit 2).

   "AFW MINIMUM FLOW REQUIREMENTS
   IF any AFW pump mini-recirc valve fails shut OR annunciator C01 A 1-9, INSTRUMENT AIR HEADER PRESSURE LOW in alarm, THEN monitor and maintain minimum AFW flow or stop the affected AFW pump as necessary to control S/G levels.
   o P-38A minimum flow - GREATER THAN 50 GPM
   o P-38B minimum flow - GREATER THAN 50 GPM
   o P-29 minimum flow - GREATER THAN 75 GPM"

2. The following step was revised in the Alarm Response Procedure , ARP C01 A 1-9, INSTRUMENT AIR HEADER PRESSURE LOW UNIT 0.

   3.1 Perform the following:
   3.1.1 IF any AFW pump mini-recirc valve fails shut in conjunction with this alarm,
   THEN monitor and maintain minimum AFW flow
   OR stop the affected AFW pump as necessary to control S/G levels.

   3.1.2 Respond per ARP C01 D 1-2, INSTRUMENT AIR HEADER PRESS LO.

Point Beach Operations Manual OM 3.7, "AOP and EOP Procedure Sets Use and Adherence," Section 4.8, "Foldout Pages" specifies the following regarding use of Foldout Pages:

- Foldout pages shall be continuously evaluated whenever the controlling procedure is in use.
- Action shall be taken after the immediate action steps are complete when a foldout page criterion is met.

I.2 Search the PBNP Final Safety Analysis Report (FSAR), pending FSAR Change Requests (FCRs) with assigned numbers, the CLB (Regulatory Commitment) Database, the Fire Protection Evaluation Report (FPER), the Technical Specifications, Technical Specifications Bases, and the Technical Requirements Manual. Identify below the pertinent design function(s), performance requirements, methods of controlling or performing design functions, design basis fission product barrier limits and methods of evaluation. Identify where the pertinent information is described in the above documents (by document section number and title).

The Auxiliary Feedwater (AFW) system has the following functions described in the licensing basis:

a. To automatically start and ensure that adequate feedwater is supplied to the steam generators for heat removal during accidents which may result in a main steam safety valve opening (Loss of Normal Feedwater – including ATWS, and Loss of AC to the Station Auxiliaries).
b. To automatically start and provide flow to maintain steam generator levels during accidents which require or result in rapid reactor coolant system cooldown (Steam Generator Tube Rupture and Rupture of a Steam Pipe).
c. To allow the isolation of all lines to the ruptured steam generator in the SGTR event.
d. To provide sufficient feedwater to remove decay heat from both units for one hour during a station blackout event (TDAFP only).
e. To provide sufficient flow to the steam generators to remove decay heat to achieve cold shutdown within 72 hours following a plant fire (Appendix R).
f. To withstand a seismic event (designed as seismic Class 1) and to ensure that steam generator levels are maintained during a seismic event.
g. To provide flow to the steam generators during plant startup and shutdown, and during hot shutdown or hot standby conditions for chemical additions and when operation of the main feedwater and condensate systems is not warranted.

FSAR 7.2.3.2 Item e. Steam Generator Water Level and Feedwater Flow states: "The basic function of the reactor protection trips associated with low steam generator water level and low feedwater flow is to preserve the steam generator heat sink for removal of long-term residual heat (See Figure 7.2-12). Should a complete loss of feedwater occur with no reactor protection action, the steam generators would boil dry and cause an overtemperature/overpressure excursion of the reactor coolant."

Reactor trips on temperature, pressure, and pressurizer water level will trip the plant before there is any damage to the core or reactor coolant system. However, the residual heat remaining after a trip would cause thermal expansion and discharge of the reactor coolant to containment through the pressurizer relief valves and pressurizer relief tank.

Redundant auxiliary feedwater pumps are provided to prevent the loss of steam generator inventory. Reactor trips act before the steam generators are dry, to reduce the required capacity and starting time requirements for the auxiliary feedwater pumps and minimize the thermal transient on the reactor coolant system and steam generators."

FSAR 7.3.3.4, "Manual AFW Flow Control During Plant Shutdown" states: "The successful operation of the engineered safety features only involves actuation, with one exception. This exception is manually controlling steam generator water level using the auxiliary feedwater pumps during plant shutdown, to remove reactor decay and sensible heat. This manual control involves positioning the auxiliary feedwater flow control valves in order to maintain proper steam generator water level. Steam generator water level indication and controls are located in the control room and at a local control station.

FSAR 10.1, Steam and Power Conversion System, Section 10.1.1 Design Basis under "Main Feedwater" states: "Reactor trip is actuated either on a coincidence of sustained steam flow - feedwater flow mismatch, coupled with low level in any steam generator or by a low-low steam generator water level. These trips are discussed in further detail in Section 7.2.

Whenever a reactor trip occurs, the main feedwater control valves move to the fully opened position to increase the feedwater flow to the steam generators for faster reduction of reactor coolant temperature to the no-load average temperature value. The

valves remain fully open until either one of the following conditions occurs, at which time the respective valve, or valves, fully close:

Abnormally high steam generator level;
Safety injection signal; or
Average temperature error signal (Between measured Tavg and the no load TREF) reduces to a preset level.

Either a high steam generator level or a safety injection signal will close the feedwater bypass valves."

FSAR Figure 10.1-2, Sheet 2 (Unit 1) and Figure 10.1-2A, Sheet 2 show that valves 1/2 CS-466 and 1/2 CS-476 (Main Feedwater Regulating Valves) and valves 1/2 CS-480 and 1/2 CS-48 (Main Feedwater Regulating Bypass Valves) are air-operated, failed closed valves.

FSAR 10.2.1, Design Basis states: "The auxiliary feedwater system is designed to supply high-pressure feedwater to the steam generators in order to maintain a water inventory for removal of heat energy from the reactor coolant system by secondary side steam release in the event of inoperability or unavailability of the main feedwater system. In order to meet the design basis required in the Loss of Normal Feedwater/Loss of All AC analysis, one motor driven auxiliary feedwater pump provides 200 gpm of flow either to one steam generator or split between two steam generators within 5 minutes following receipt of a low-low steam generator water level setpoint signal. Redundant supplies are provided by two pumping systems using different sources of power for the pumps. The design capacity of each system is set so that the steam generators will not boil dry nor will the primary side relieve fluid through the pressurizer relief valves, following a loss of main feedwater flow with a reactor trip."

FSAR 10.2.2, System Design and Operation, states: "The auxiliary feedwater system consists of two electric motor-driven pumps, two steam turbine-driven pumps, pump suction and discharge piping, and the controls and instrumentation necessary for operation of the system. Redundancy is provided by utilizing two pumping systems, two different sources of power for the pumps, and two sources of water supply to the pumps. The system is categorized as seismic Class I and is designed to ensure that a single fault will not obstruct the system function."

FSAR 10.2.2 also states:

- For the motor-driven pumps: "Each pump has an AOV, AF-4007 for P-38A and AF-4014 for P-38B, controlled recirculation line back to the condensate storage tanks to ensure minimum flow to prevent hydraulic instabilities and dissipate pump heat."

- For the turbine-driven pumps: "Each pump has an AOV (AF-4002) controlled recirculation line back to the condensate storage tanks to ensure minimum flow to dissipate pump heat."

FSAR 10.2.2 also states:

During normal plant operations, the auxiliary feedwater system is maintained in a standby condition ready to be placed in operation automatically when conditions require. The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

Turbine-driven feedwater pumps

1. Low-low water level in both steam generators in one unit starts the corresponding pump.

2. Loss of both 4.16 kv buses supplying the main feedwater pump motors in one unit starts the corresponding auxiliary feedwater pump.

3. Trip or shutdown of both main feedwater pumps or closure of both feedwater regulating valves in one unit starts the corresponding pump. These signals are processed through AMSAC at power levels above 40%.

Motor-driven feedwater pumps

1. Low-low water level in either associated steam generator.

2. Trip or shutdown of both main feedwater pumps or closure of both feedwater regulating valves in one unit. These signals are processed through AMSAC at power levels above 40%.

3. Safeguards sequence signal.

FSAR Figure 10.2-1 shows that 1/2AF-4002 for 1/2P-29 (turbine-driven pumps), and AF-4007 for P-38A, AF-4014 for P-38B (motor-driven pumps) are all fail-closed valves.

CLB References:
FSAR 7.2.3.2- Specific Control and Protection Interactions
FSAR 7.3.3.4 - Manual AFW Flow Control During Plant Shutdown
FSAR Section 7.4.1 – AMSAC
FSAR Section 10.1 - Steam and Power Conversion System
FSAR Section 10.2 – Auxiliary Feedwater
FSAR Figure 10.2-1 Sheet 1 – Bech M-217 Sh. 1 – Auxiliary Feedwater System
FSAR Figure 10.2-1 Sheet 2 – Bech M-217 Sh. 2 – Auxiliary Feedwater System
FSAR Section 14.1.10 – Loss of Normal Feedwater
FSAR Section 14.1.11 – Loss of All AC Power to the Station Auxiliaries
FSAR Section 14.2.4 – Steam Generator Tube Rupture
FSAR Section 14.2.5 – Rupture of a Steam Pipe
FSAR Appendix A.1 – Station Blackout
FPER 5.2.2 – Safe Shutdown Systems and Equipment
FPER 5.2.5.2.3 – Auxiliary Feedwater Pump Room
FPER 6.6 - Safe Shutdown Systems
FPER 6.6.4 - Auxiliary Feedwater System
Tech Spec 3.7.5 – Auxiliary Feedwater
Tech Spec Bases B 3.7.5 – Auxiliary Feedwater

PART II –DETERMINE IF TECHNICAL SPECIFICATION CHANGE REQUIRED

Does the proposed activity involve a change to any Technical Specification? Changes to Technical Specifications require a License Amendment Request.

Technical Specification Change REQUIRED: ☐ YES ☒ NO

If a Technical Specification or change is required, explain what the change should be and why it is required.

PART III – 10 CFR 50.59 EVALUATION

NOTE: **A unique written basis is required to answer each question below. The "NMC 10 CFR 50.59 Resource Manual" (RESOURCE MANUAL) should be used to determine the content of each response. Identify references used to perform evaluation at the beginning of the form and within the written responses.**

III.1. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the CLB? (See Section 6.2.1 of the RESOURCE MANUAL) ☐ YES ☒ NO

Basis for answer:

The auxiliary feedwater (AFW) system responds to plant transients and other initiating events or accidents. Failure of an AFW pump to start or run does not initiate a transient or an accident.

The AFW pumps support the following functions:
- To automatically start and ensure that adequate feedwater is supplied to the steam generators for heat removal during accidents which may result in a main steam safety valve opening (Loss of Normal Feedwater – including ATWS, and Loss of AC to the Station Auxiliaries).

- To automatically start and provide flow to maintain steam generator levels during accidents which require or result in rapid reactor coolant system cooldown (Steam Generator Tube Rupture and Rupture of a Steam Pipe).
- To provide sufficient feedwater to remove decay heat from both units for one hour during a station blackout event (TDAFP only).
- To provide sufficient flow to the steam generators to remove decay heat to achieve cold shutdown within 72 hours following a plant fire (Appendix R).

The permanent procedure changes discussed above provide directions to operators to prevent AFW pump failure in the event of a loss of instrument air. Manual operator action to control steam generator level is already part of the Point Beach licensing basis as discussed in Section I.1. In order to accomplish this, operators would control flow to the steam generators by discharge valves, or by securing AFW pumps after decay heat rates decrease and sensible heat is removed from the reactor coolant system. With the exception of an ATWS event, the FSAR Chapter 14 analyses demonstrate that only one pump is required (i.e., 200 gpm flow) to meet the acceptance criteria for Loss of Normal Feedwater and Loss of AC to the Station Auxiliaries. In an ATWS event the analyses do not credit a reactor trip, therefore a larger AFW flow is required to mitigate the event because of continuing nuclear power generation.

The permanent procedure changes ensure adequate flow is maintained to the steam generators, thus providing sufficient cooling to the reactor coolant system to prevent overfill of the pressurizer and the possibility of a small-break LOCA due a relief valve failing due to water relief. Therefore, the frequency of occurrence of a small LOCA that may result from a stuck open pressurizer PORV is not increased.

III.2. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the CLB? (See Section 6.2.2 of the RESOURCE MANUAL)
☐ YES ☒ NO

Basis for answer:

An AFW pump failing to start is considered in the FSAR. The procedure changes prescribe that the operator maintain minimum AFW flow or stop the affected AFW pump as necessary to control S/G levels. Operation of AFW pumps at reduced flows and their ability to sustain repeated starts is a concern.

The procedure flow requirements are based on Flowserve Corporation Pump Division letter dated March 2, 2001, Subject: Aux Feed Water Pumps Minimum Flow Analysis. The letter states that the turbine-driven aux feed pump (TDAFP) can operate at a flow rate of 75 gpm for up to 60 hours. These operating flows and times are provided only for determining recommended frequency of overhaul and do not represent time to failure. (Prior to OEM recommended overhaul, the TDAFP can operate up to 1500 hours at a flowrate of 130 gpm, and an unlimited amount of time at 210 gpm.) The letter states that the motor-driven aux feed pump (MDAFP) can operate at a flow rate of 50 gpm for up to 60 hours. (The MDAFP can operate up to 1500 hours at a flowrate of 75 gpm, and an unlimited amount of time at 105 gpm.) Time to failure would be significantly longer times and lower flows.

The licensing basis allows the use of manual operator control of the SG level. Although not explicitly stated in the FSAR, manual control includes starting and stopping the pumps.

NEMA guidelines state that motor driven pumps may be started twice from an initial standby condition with no restrictions. After the initial starts, NEMA MG-1 recommends subsequent starts after "... all conditions affecting operation have been thoroughly investigated and the apparatus has been examined for evidence of excessive heating." This guidance encompasses situations where the load or prime mover may have experienced a fault which causes a failure to start or run; excessive temperatures are not expected with the equipment functioning normally. Additional starts, therefore are allowed and are not expected to cause machine failure. Furthermore, motor starting nameplate information directs that after initial starts, the motor may be restarted if run time has exceeded 15 minutes or it has been secured for 60 minutes. Excessive consecutive starting will decrease overall motor lifetime. Starting duty limitations are provided by the manufacturer to provide the longest possible motor lifetime.

At least a minimum of 3 starts of the motor driven auxiliary feedwater pumps is allowed in the first 15 minutes of operation. An additional one or two starts is allowed provided run time is at least 15 minutes between these later starts.

This is in excess of that necessary to control auxiliary feedwater. After starting a secured AFW pump, the operator would throttle flow to maintain SG level. By throttling flow, operators would control level and it would not be necessary to stop the pump nor start the pump a third time. Typical auxiliary feedwater flow rates during the time following the reactor trip would be sufficient to provide flow to ensure pump operability. Based on operator training and knowledge, only one pump would be started if operators had secured all the pumps to control flow.

There are no starting cycle limitations on the turbine-driven AFW pumps. Governor and turbine vendors were contacted by the AFW system engineer to ensure that no duty cycle concerns for the turbine exist. The motor-operated turbine steam supply valves are rated for 5 minutes of operation every hour which corresponds to five open/shut cycles. The nameplate guidance for motors and valves is based on manufacturers' recommendations for long life and are considered conservative.

The MDAFP pump motors are supplied from Westinghouse DB-50 Air Circuit breakers. Vendor information (Westinghouse Application Data AD 33-760, dated November 1967) indicates an In-rush, Non-fault Duty Cycle of 750 breaker operations. The recommended operating conditions are a frequency of operation not to exceed 20 in 10 minutes or 30 in one hour. Since MDAFP are only used during start-up, shutdown, and required system testing, the number of operations are well below these breaker operating limits. If an operator is required to secure and restart pumps during a transient, the actions would be performed at a frequency of operation much less than the recommended frequency of operation per hour. The total number of breaker operations would also be less than the recommended value for the entire transient.

The remaining possible malfunction is the operator failing to maintain minimum AFW flow as required to prevent pump failure (i.e, reduce flow through the pump below the minimum value), or failing to secure the pump. The AFW minimum flow requirements are provided on the EOP foldout pages. OM 3.7 requires that foldout pages shall be continuously evaluated whenever the controlling procedure is in use. Immediately after the issue regarding the mini-recirc valves was identified in CR 01-3595, on-shift and on-coming operating crews were briefed on the issue, and just-in-time training was provided in the simulator to subsequent on-coming crews. The control room panels provide AFW flowrate indication for each of the four AFW pumps. There are procedures, training, and existing individual AFW pump flowrate indications in place in the control room to ensure the operator performs this function.

Based on the above, and given the fact that the licensing basis has demonstrated that only one pump (i.e., 200 gpm) is required (per unit) to function to mitigate the applicable transients, the licensing basis will be met in that one AFW pump will still be available to each unit. The MDAFPs and the TDAFPs will be operated within the required service conditions and minimum flows prescribed in operating procedures and reinforced by operator training. Therefore, there is not more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the CLB.

III.3. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the CLB? (See Section 6.2.3 of the RESOURCE MANUAL.) ☐ YES ☒ NO

Basis for answer:

The FSAR accident analysis assumes a minimal AFW flow of 200 gpm, which is within the capability of one AFW pump at pressures greater than the peak calculated SG pressure in the analysis. Therefore the required minimum AFW flow capability will continue to be met by implementation of these procedure changes.

The FSAR analysis of the loss of external electrical load event has demonstrated that the acceptance limits for the RCS and steam generator secondary pressure, and for departure from nucleate boiling ratio (DNBR), have been met. Based on these results, the pressure boundary integrity of the RCS and the secondary side are not challenged, and the fuel cladding integrity is also unchallenged.

The FSAR analysis of the loss of normal feedwater and the loss of ac power events has demonstrated that the pressurizer volume remains below the acceptance limit (i.e., the pressurizer does not overfill). Therefore, the consequences of these events are analyzed in that no water relief occurs and no challenge to the pressurizer PORV or safety valves is created.

The FSAR Steam Generator Tube Rupture (SGTR) and Rupture of a Steam Pipe radiological analyses credit auxiliary feedwater for cooldown through use of the intact steam generator. No specific flowrates are specified in the FSAR, however based on mass release values in the fuel upgrade RTSR, it is determined that the flow rate of 133 gpm is required over the 24 hour cooldown period. The AOP for loss of instrument air directs the operators to gag the recirc valve only to ensure

minimum flow. A caution alerts the operator that gagging could potentially divert flow. During the SGTR or rupture of a steam pipe accident, the Emergency Operating Procedures ensure that auxiliary feedwater is supplied to the non-affected generator and gagging the recirculation valves is not necessary until after cooldown. The flowrates required during these scenarios ensure that the pumps are not deadheaded. The radioactive release from the intact generator is modeled as the amount of steam release required to remove decay heat until RHR cut-in conditions are reached. Since the procedure revisions have been implemented to ensure AFW flow is available, then there is no change in the radiological consequences of these accidents.

Based on the above, the proposed activity does not result in any increase in the consequences of accidents evaluated in the CLB.

III.4. Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the CLB? (See Section 6.2.4 of the RESOURCE MANUAL) ☐ YES ☒ NO

Basis for answer:

As indicated in the answer to question III.3, as long as an AFW flowrate required for cooldown can be maintained by the operators by using the procedure changes, then there will be no impact on radiological consequences. The discussion under question II.2 demonstrates that the likelihood of increased failures of AFW pumps is not more than ~~a~~ minimal. Since the procedure changes ensure that the AFW flowrate can be maintained, ~~then~~ there is no increase in the consequences of an SSC important to safety previously evaluated.

III.5. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the CLB? (See Section 6.2.5 of the RESOURCE MANUAL.) ☐ YES ☒ NO

Basis for answer:

The auxiliary feedwater (AFW) system responds to plant transients and other initiating events or accidents. Failure of an AFW pump to start or run does not initiate a transient or an accident. However, if all AFW pumps fail to start or fail to run, then this would be an accident of a different type. If all AFW pumps start and run in response to a transient, then the steam generators could be overfilled or the RCS overcooled unless operator action is taken to reduce AFW flow. There are some plant design functions which could contribute to overcooling the RCS or overfilling the steam generators by providing additional feedwater to the steam generator inventory. FSAR 10.1 states: "Whenever a reactor trip occurs, the main feedwater control valves move to the fully opened position to increase the feedwater flow to the steam generators for faster reduction of reactor coolant temperature to the no-load average temperature value. The valves remain fully open until either one of the following conditions occurs, at which time the respective valve, or valves, fully close:

Abnormally high steam generator level;
Safety injection signal; or
Average temperature error signal (Between measured Tavg and the no load TREF) reduces to a preset level."

Given a loss of instrument air, the main feedwater regulating and bypass valves fail close. Therefore the contribution of this effect to overcooling the RCS and overfilling the SGs will be less than for a normal reactor trip. However, closure of the regulating valves will cause all AFW pumps, to start as discussed in FSAR 10.2.2. So operator action is required to control AFW flow if all pumps start as designed. If the design basis event occurs and only one AFW pump is available per unit, operators would control flow at 200 gpm until steam generator is within the desired bands listed in the EOPs. So initial failure of one or two AFW pumps to start would reduce the requirements for operators to manually control AFW flow to prevent pump failure. This case does not create the possibility that an accident of a different type will occur.

For the situation of all AFW pumps starting as designed, it is likely that the motor driven pumps will feed the unit with the lower steam pressure (due to variations in actual SG relief valve settings). In this case the unit specific TDAFP will be feeding its respective unit's steam generators, while the two MDAFPs will feed the opposite unit along with its unit-specific TDAFP. The latter unit would be the one at risk for overcooling the RCS or overfilling the SGs, and AFW flow would have to be reduced. In this scenario it is very unlikely that the operator would cause all three of these AFPs to fail. The operators have been provided procedural guidance and training to address this scenario as discussed in III.2. Even if operator error caused the failure of two pumps, then there is still one AFW pump available per unit to perform their design basis function. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the CLB

III.6. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in CLB? (See Section 6.2.6 of the RESOURCE MANUAL) ☐ YES ☒ NO

Basis for answer:

The FSAR considers the malfunction of an AFW pump failing to start. One situation considered in this evaluation is the possibility of running AFW pumps failing to run after starting because a minimum flow for pump cooling is not maintained due to the loss of instrument air to the pump mini-recirculation valves, and an operator failing to secure pump(s), or failing to maintain minimum pump flow required for cooling. The other situation is the pumps failing to restart after having been secured by an operator. In either of these situations, the result of the malfunction is that the AFW pump fails to provide feedwater to the steam generators. Therefore, the proposed activity does not create the possibility of a malfunction with a different result. The possibility of all AFW pumps failing is discussed in III.5 as the possibility of an accident of a different type.

III.7. Does the proposed activity result in a design basis limit for a fission product barrier as described in the CLB being exceeded or altered? (See Section 6.2.7 of the RESOURCE MANUAL) ☐ YES ☒ NO

Basis for answer:

The procedure changes will provide operators direction such that an AFW flow of 200 gpm will be available. The FSAR analysis of the loss of external electrical load event demonstrates that the acceptance limits for the RCS and steam generator secondary pressure, and for departure from nucleate boiling ratio (DNBR), have been met. Based on these results, the pressure boundary integrity of the RCS and the secondary side are not challenged, and the fuel cladding integrity is also unchallenged.

The FSAR analysis of the loss of normal feedwater and the loss of ac power events has demonstrated that the pressurizer volume remains below the acceptance limit (i.e., the pressurizer does not overfill). The consequences of these events are analyzed in that no water relief occurs and no challenge to the pressurizer PORV or safety valves is created. Therefore no design basis limit for a fission product barrier is exceeded or altered.

III.8. Does the proposed activity result in a departure from a method of evaluation described in the CLB used in establishing the design bases or in the safety analyses? (See Section 6.2.8 of the RESOURCE MANUAL) ☐ YES ☒ NO

Basis for answer:

The procedure changes do not involve a method of evaluation as defined in the NMC 50.59 Resource Manual. Therefore the procedure changes do not result in a departure from a method of evaluation as described in the CLB used in establishing the design bases or in the safety analyses.

PART IV – 10 CFR 50.59 EVALUATION CONCLUSION

NOTE: If the answer to any of the eight 10 CFR 50.59 evaluation questions above is "YES", then the proposed activity may NOT be implemented until a License Amendment Request has been submitted to and approved by the NRC.

Based on the results of this evaluation, check ONE of the following:

☒ NO activity requiring prior NRC approval per 10 CFR 50.59 was identified AND NO Technical Specification change is involved. This activity may be implemented in accordance with applicable procedures.

☐ An activity requiring prior NRC approval per 10 CFR 50.59 WAS identified. This activity cannot be implemented without a license amendment request and NRC approval.

☐ A Technical Specification change IS involved with the proposed activity, but NO activity requiring prior NRC approval per 10 CFR 50.59 was identified. This activity cannot be implemented without a license amendment request and NRC approval.

☐ A Technical Specification change IS involved with the proposed activity, AND an activity requiring prior NRC approval per 10 CFR 50.59 was identified. This activity cannot be implemented without a license amendment request and NRC approval.

PART V – LICENSING BASIS UPDATE

Document whether an update to the licensing basis is required as a result of the activity and this evaluation. Mark ALL that apply. If no changes are required, then PART V is ☐ NOT APPLICABLE.

☐ A Regulatory Commitment (CLB Database) change is REQUIRED. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

☒ A Point Beach FSAR change is REQUIRED. If an FSAR change is required, then initiate an FSAR Change Request (FCR) per NP 5.2.6. The prepared/reviewed FCR is to be included with the evaluation provided for MSS review.

☒ A Technical Specification Bases change is REQUIRED. If a change to the Technical Specification Bases is required, then initiate a Technical Specification Bases change per NP 5.2.15.

☐ A Technical Requirements Manual change is REQUIRED. If a change to the Technical Requirements Manual is required, then initiate a Technical Requirements Manual change per NP 5.2.15.

PART VI – 10 CFR 50.59 EVALUATION SUMMARY

The evaluation summary is included in the Annual Results and Data Report (NP 5.2.3) to meet the requirements of 10 CFR 50.59(d)(2). The evaluation summary contains three concise paragraphs: the activity description, a summary of the evaluation basis for answering the 10 CFR 50.59 questions, and the conclusion whether a license amendment request and/or Technical Specification change is required. No new or different information from that in the evaluation (Parts I – IV) is to be included in the summary.

Activity Description: The proposed activity is to make permanent procedure changes that were implemented in response to a condition that was identified where, with a procedure-directed operator action to control steam generator level (which could be accomplished by reducing flow through one or more AFW pumps), concurrent with a loss of instrument air (which would cause the AFW pumps' mini-recirculation valves to fail close), the potential existed for a simultaneous failure of the multi-stage high pressure AFW pumps due to very low or no flow through running AFW pumps. This 50.59 evaluation will review the procedure changes as a permanent change to the procedures as described in the FSAR to confirm consistency with the licensing basis. The permanent procedure changes will restore the AFW pumps to fully operable status.

Summary of Evaluation: The auxiliary feedwater (AFW) system responds to plant transients and other initiating events or accidents. Failure of an AFW pump to start or run does not initiate a transient or an accident. The permanent procedure changes ensure adequate flow is maintained to the steam generators, thus providing sufficient cooling to the reactor coolant system to prevent overfill of the pressurizer and the possibility of a small-break LOCA due a relief valve failing due to water relief. Therefore, the frequency of occurrence of a small LOCA that may result from a stuck open pressurizer PORV is not increased. Operation of AFW pumps at reduced flows, and their ability to sustain repeated starts were considered. The minimum flows in the procedures meet vendor recommendations. Impact of periodic restarting of AFW pumps, motors and turbines was considered, along with electrical breaker and MOV operation. Operating procedures and operator training has addressed operator action to maintain minimum AFW flows or to secure the pump(s). Therefore, there is not more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the CLB. The FSAR accident analysis assumes a minimal AFW flow of 200 gpm, which is within the capability of one AFW pump at pressures greater than the peak calculated SG pressure in the analysis. The required minimum AFW flow capability will continue to be met by implementation of these procedure changes. Therefore there is no increase in radiological consequences due to accidents or malfunctions of SSC important to safety. Even if operator error caused the failure of two pumps, then there is still one AFW pump available per unit to perform their design basis function. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the FSAR. The result of the malfunction of a pump, due to failing to restart or

failing due to low flow, is that the AFW pump fails to provide feedwater to the steam generators which is the same result. Therefore, the proposed activity does not create the possibility of a malfunction with a different result. The FSAR analysis of the loss of normal feedwater and the loss of ac power events has demonstrated that the pressurizer volume remains below the acceptance limit (i.e., the pressurizer does not overfill). The consequences of these events are analyzed in that no water relief occurs and no challenge to the pressurizer PORV or safety valves is created. Therefore no design basis limit for a fission product barrier is exceeded or altered. The procedure changes do not involve a method of evaluation, therefore the procedure changes do not result in a departure from a method of evaluation used in establishing the design bases or in the safety analyses.

Conclusion: The conclusion of the 50.59evaluation was that prior NRC approval is not required to permanently implement these procedure changes.

April 27, 2002
11:00pm - 12:00pm

Members: Vito Kaminskas (Chairman)
Stu Thomas (RP) Tom Carter (ENG) Jack Gadzala (LIC)
Rick Arnold (CHEM)

* Part Time
Visitors/Guests:
Rick Wood
James Hanna

1) In accordance with FSAR 1.4.19.8.d and 1.4.19.9.c the MSS reviewed the following safety evaluations and found no unreviewed safety questions:

EVAL 2002-005, Permanent Procedure Changes to Address Potential for Simultaneous Failure of All AFW Pumps

This evaluation is being brought to staff with changes.

- Staff discussion regarding issues brought up at the previous MSSM on this 50.59 evaluation.
- Staff discussion of the changes to address previous concerns that had been brought up by previous MSSM.

Staff Approved.
After the approval from staff, Rick Wood made a few minor changes.

- The last sentence on page one of the safety evaluation refers to the backup nitrogen as NON safety-related. The system has now been designated as safety-related and therefore, this last sentence will be removed. The removal of the sentence has no impact on the conclusion of the evaluation
- Section III.4, Second sentence, strike the second to last word, "a" for clarity.
- Section III.4, Last sentence should read, "since the procedure changes ensure that the AFW flowrate can be maintained, there is no increase in the consequences of a malfunction of an SCC important to safety previously evaluated.

These changes were made prior to the Plant Managers signature.

K. M. Locke

(Approved)
Vito Kaminskas

cc: JOSRC c/o L. Schofield
K. M. Locke
D. B. Black
M. Warner
E. J. Weinkam
J. J. Walsh
File