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10 CFR 50.90

1CAN102202

October 31, 2022

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Application to Remove Technical Specification Condition Allowing Two Reactor Coolant Pump Operation

Arkansas Nuclear One, Unit 1
NRC Docket No. 50-313
Renewed Facility Operating License No. DPR-51

References:

1. NUREG-1430, "Standard Technical Specifications – Babcock and Wilcox Plants," Revision 5, Volume 1, (ADAMS Accession No. ML21272A363), dated September 2021

2. Nuclear Energy Institute (NEI), NEI 15-03, Revision 3, "Licensee Actions to Address Nonconservative Technical Specifications," (ADAMS Accession No. ML20100G899), dated March 2020

As required by 10 CFR 50.90, Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1).

Entergy hereby requests a change to the ANO-1 TSs to remove the condition that allows for two reactor coolant pump (RCP) operation while at power with one RCP in operation in each Reactor Coolant System (RCS) loop. The change removes TS 3.4.4 "RCS Loops – MODES 1 and 2" Condition A, "One RCP not in operation in each loop," Required Action A.1 to "Restore one non-operating RCP to operation," and its Completion Time of "18 hours." This change makes ANO-1 TS 3.4.4 consistent with the U.S. Nuclear Regulatory Commission (NRC) Standard Technical Specifications for Babcock and Wilcox (B&W) Plants (Reference 1).

Entergy identified that the current methodology for calculating limits with four or three operating RCPs could not be clearly traced to the method originally used to calculate limits with one RCP per loop operating, and that operation in future fuel cycles could not be assured to remain within limits under all conditions.

The proposed change corrects the potentially non-conservative TS 3.4.4 Condition A, which allows power operation with one RCP in service per loop for up to 18 hours before entering Condition B. Eliminating Condition A and defaulting to the current Condition B, which requires ANO-1 be in Mode 3 within the next 6 hours, removes the potential non-conservatism by forcing a plant shutdown if a third RCP cannot be placed back into service within the 6 hours allowed for shutdown. The proposed change ensures system operability requirements are consistent with safety analyses. This issue is being tracked by the Entergy corrective action program (CAP), with compensatory actions specified in an operations standing order which prohibits operation in LCO 3.4.4 Condition A without also immediately entering Condition B, "Be in Mode 3 within 6 hours." This standing order will remain in effect until the license is amended. This satisfies the guidance in Nuclear Energy Institute (NEI) 15-03, Revision 3, "Licensee Actions to Address Nonconservative Technical Specifications" (Reference 2) which is endorsed by Regulatory Guide 1.239.

The enclosure provides a description and assessment of the proposed change. In addition, the enclosure concludes that the proposed amendment does not involve a significant hazards consideration. Attachment 1 of the enclosure provides the existing TS pages marked to show the proposed change. Attachment 2 of the enclosure provides a markup of the current TS Bases pages associated with this change and are provided for information only. Attachment 3 of the enclosure provides revised (clean) TS pages.

This letter contains no new regulatory commitments.

Approval of the proposed amendment is requested by November 30, 2023. Once approved, the amendment shall be implemented within 90 days.

In accordance with 10 CFR 50.91, Entergy is notifying the State of Arkansas of this amendment request by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Riley Keele, Manager, Regulatory Assurance, Arkansas Nuclear One, at (479) 858-7826.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on October 31, 2022.

Respectfully,

Philip Couture

Digitally signed by Philip Couture
Date: 2022.10.31 11:44:44 -05'00'

Phil Couture

PC/mar

Enclosure: Evaluation of the Proposed Change

Attachments to Enclosure:

1. Technical Specification Page Markup
2. Technical Specification Bases Page Markups (Information Only)
3. Retyped Technical Specification Page

cc: NRC Region IV Regional Administrator

NRC Senior Resident Inspector – Arkansas Nuclear One

NRC Project Manager – Arkansas Nuclear One

Designated Arkansas State Official

Enclosure

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Evaluation of the Proposed Change

Evaluation of the Proposed Change

1.0 SUMMARY DESCRIPTION

Entergy Operations, Inc. (Entergy) requests U.S. Nuclear Regulatory Commission (NRC) review and approval of a proposed amendment to the Arkansas Nuclear One, Unit 1 (ANO-1) Renewed Facility Operating License DPR-51, Appendix A, Technical Specifications (TSs), to revise Technical Specification (TS) 3.4.4, "Reactor Coolant System (RCS) Loops – MODES 1 and 2" to eliminate Condition A which allows one reactor coolant pump (RCP) in each loop to be out of service for up to 18 hours. Since this condition would no longer be permitted, an analysis was performed to determine if the TS 3.3.1 "Reactor Protection System (RPS) Instrumentation" Table 3.3.1-1 Function 7, "Reactor Coolant Pump to Power" Allowable Value could remain at its current value of "≤ 55% rated thermal power (RTP) with one pump operating in each loop," or if a change to the power-to-pumps monitor trip setpoint would be required. The power-to-pumps trip, also referred to as the pump-power-monitor (PPM) trip, ensures a reactor trip when no reactor coolant pumps are operating on one steam generator loop and provides a backup trip to prevent departure from nucleate boiling (DNB) if reactor power is too high with only one RCP operating per loop. The analysis determined that it was acceptable for the PPM trip setpoint to remain set at ≤ 55% RTP with one RCP operating per loop.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Reactor Coolant System Overview

The primary function of the RCS is to remove the heat generated in the fuel and transfer this heat via the steam generators (SGs) to the secondary plant.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two RCPs. An RCP is located in each of the SG cold legs. The RCP flow rate has been sized to provide core heat removal, with appropriate margin to departure from nucleate boiling (DNB), during power operation and for anticipated transients originating from power operation. TS 3.4.4 currently requires two RCS loops with either three or four pumps to be in operation. With three RCPs in operation, the reactor power level is restricted to 75% RTP to preserve the core power-to-flow relationship, thus maintaining the margin to DNB. The intent of the TS is to require core heat removal with forced flow through both loops during power operation. Specifying the minimum number of RCPs is an effective technique for designating the proper forced flow rate for heat transport and specifying two loops provides for the needed amount of heat removal capability for allowed power levels.

The RPS initiates a reactor trip, if necessary, to protect against violating the core fuel design limits and to protect the RCS pressure boundary during abnormalities. By tripping the reactor, the RPS also assists the Engineered Safety Feature (ESF) systems in mitigating accidents.

The limiting safety system settings (LSSS), defined in the TS 3.3.1 basis and listed as the ALLOWABLE VALUE column in Table 3.3.1-1, in conjunction with the Limiting Conditions of Operation (LCOs) and administrative controls, establish the threshold for protective system

action to prevent exceeding specified acceptable limits during Design Basis Accidents (DBAs). Acceptable consequences for accidents are that the offsite dose shall be maintained within 10 CFR 50.67 accident source term limits or other limits approved by the NRC.

During abnormalities, one or more of the following limits are maintained:

a. For accidents other than locked rotor, the departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value. For the locked rotor accident, the minimum DNBR shall not be less than the applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions;

b. Fuel centerline temperature shall be maintained below the SL value;

c. The RCS pressure SL of 2750 psig shall not be exceeded; and

d. Reactor power shall not exceed 112% RTP.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50.67 criteria during abnormalities.

Reactor Protection System Overview

The RPS consists of four separate, redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, reactor outlet temperature, RCP status, Reactor Building (RB) pressure, main feedwater (MFW) pump turbine status, and main turbine status.

A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass the equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS) - Reactor Trip Module (RTM)," and LCO 3.3.4, "Control Rod Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of trip signals in any two of the four RPS channels will result in tripping of the reactor.

The RPS consists of four independent protection channels, each containing an RTM. Each RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels trip, the RTM in each channel actuates to remove 120 VAC power from its associated CRD trip breaker.

The reactor is tripped by opening circuit breakers and de-energizing Electronic Trip Assembly (ETA) relays that interrupt the control power supply to the CRDs. A one-out-of-two taken twice logic is used to interrupt power to the CRDs.

The RPS receives inputs from the instrumentation channels listed in TS Table 3.3.1-1. The instruments which support Function 7 compare the number of running RCPs against the current RTP (PPM trip) and generate a trip signal if one of its setpoint combinations are exceeded.

Reactor Coolant Pump Power Monitoring

RCP power monitors are inputs to the reactor PPM trip, (Refer to TS Table 3.3.1-1 "Reactor Protection System Instrumentation," Function 7). Each RCP's operating current is measured by a current transformer, which provides the current input to the associated RCP underpower relay. The voltage is measured by a potential transformer which provides the voltage input to the associated RCP underpower relays. Each RCP underpower relay provides individual RCP status to each protection channel. The current Allowable Value for one RCP in operation per loop is ≤ 55% RTP.

The RPS PPM trip setpoint is described in BAW-10179P-A "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses – Topical Report," Revision 9 (Reference 3). It is designed to allow the plant a chance to run back power without a reactor trip following a single reactor coolant pump trip and coastdown from four RCP initial conditions. As such, the PPM trip does not place any requirement on the allowed power level for four or three RCP operation, allowing the RPS high flux and flux/flow trips to maintain the required protection.

The PPM trip ensures a reactor trip when there are no reactor coolant pumps operating in one steam generator loop (0/0, 1/0, 0/1, 2/0, or 0/2 operation). This trip also provides the primary protection for the following events:

1. Multiple reactor coolant pump coastdowns.
2. Single reactor coolant pump coastdown from partial pump operation.
3. Reactor coolant pump coastdowns resulting in the loss of both pumps in either loop.

The PPM trip is designed to operate in a nearly binary manner: power operation is allowed or not allowed. The determination is based on the comparison of the measured neutron power to the allowed power for the in-service pump combination. The pump monitor in the trip string rapidly determines allowed power level for the pump combination and the bistable then determines the need for a reactor trip.

2.2 Current TS Requirements

Currently, ANO-1 TS 3.4.4, "RCS Loops – MODES 1 and 2," LCO states that in Modes 1 and 2, two RCS loops shall be in operation, with either four RCPs operating; or three RCPs operating and THERMAL POWER restricted as specified in the Core Operating Limits Report (COLR). Condition A states that with one RCP not in operation in each loop, one non-operating RCP shall be restored to operation within 18 hours or Condition B shall be entered, which requires the plant to be in Mode 3 within 6 hours if the requirements of the LCO are not met.

ANO-1 TS 3.3.1 Table 3.3.1-1 Function 7 states that the Reactor Coolant Pump to Power setpoint shall be ≤ 55% RTP with one pump operating in each loop while in Modes 1 and 2.

2.3 Reason for the Proposed Change

Framatome notified Entergy that the methodology for calculating the four and three RCP steady-state operating limits in Modes 1 and 2 could not be used in the case where only two RCPs (one RCP per loop) were in operation (Reference 10). The methodology utilized to establish the operating limits for two RCP operations is in BAW-10103A, "Emergency Core Cooling System (ECCS) Analysis of Babcock and Wilcox's 177-Fuel Assembly (FA) Lowered Loop Nuclear Steam Supply (NSS)," Revision 3 (Reference 4); however, the current reload topical report is BAW-10179P-A, Revision 9 (Reference 3), which no longer directly references BAW-10103A. Entergy has determined that this methodology may not be acceptable under all conditions for future operating cycles; therefore, TS 3.4.4 is a potentially non-conservative TS. Because of this, the guidance in Nuclear Energy Institute (NEI) 15-03, Revision 3, "Licensee Actions to Address Nonconservative Technical Specifications" (Reference 11), endorsed by Reg. Guide 1.239,"Licensee Actions to Address Nonconservative Technical Specifications" is being applied in accordance with Entergy procedures. Currently, ANO-1 is in Refueling Outage 1R30, and Operations has implemented a standing order that requires a plant shutdown to Mode 3 be performed within 6 hours if only one RCP is in operation in each RCS loop, despite the current TS allowing 18 additional hours of operation in this condition. This standing order accomplishes the same actions as the TS change proposed by this license amendment request and will remain in effect until this amendment request is approved by the NRC.

The computer code used to develop the core operating limits in BAW-10179P-A (Reference 3) only has provisions to calculate scenarios with four or three RCPs in operation, and Entergy has only been provided RPS limits for four-RCP and three-RCP operation. The non-loss-of-coolant accident (LOCA) safety analyst has been using these four-RCP and three-RCP limits to develop the RPS power/imbalance/flow allowable value setpoints and protective limits, which include limits for two-RCP setpoints that are estimated based on the four-RCP and three-RCP setpoints. Entergy performed an evaluation of the two-RCP steady state DNB limits for Cycles 29 and 30 and determined that acceptable two-RCP RPS power/imbalance/flow setpoints could be justified for Cycles 29 and 30 if part of the available DNB thermal margin was used. Additionally, an analysis was performed to demonstrate that the Allowable Value for the power-to-pumps trip could remain at its current value of 55% with one RCP operating in each loop. Entergy also determined that it was acceptable to operate in Cycle 31 with the same Operations standing order and corrective action plan that was in place for Cycle 30 without any additional limitations. Once the proposed TS change is approved and the TS are updated, the Operations standing order will be discontinued.

From BAW-10103A (Reference 5):

> *"To allow an operating configuration with less than four RCPs on the line (partial loop), the staff requires an analysis of the predicted consequences of a LOCA occurring during the proposed partial loop operating mode(s)."*

BAW-10103A (Reference 5) states that Babcock and Wilcox (B&W) had submitted a generic analysis for three-RCP operation, and that assuming an RCS break at the worst location at an initial power of 77% RTP, cladding temperature was within 10 CFR 50.46 "Acceptance Criteria for ECCS" limits based on time-in-life sensitivity studies.

Although the current TS 3.4.4 limits indefinite operation to a minimum of three RCPs in total, design evaluation (including analyses at steady state, ECCS initial conditions, and DNB

conditions) previously determined that operation with one RCP operating in each loop (two RCPs total) was acceptable when core thermal power was restricted to being proportionate to RCS flow. However, continued power operation with two RCPs removed from service was restricted to 24 hours total since not all transient and accident conditions were analyzed for this condition.

From BAW-10103A:

> *"Since an analysis of ECCS cooling performance with one idle RCP in each loop has not been submitted, power operation in this configuration will be limited in 177-FA plants with lowered loops to 24 hours."*

BAW-10103A had previously been used to determine the acceptability of two-RCP operation; however, this topical report has not been directly referenced in BAW-10179P-A since approximately the time that ANO-1 transitioned to using Mark-B-HTP fuel in Cycle 20, which started in December 2005. Due to the limitations associated with the computer code used to perform partial loop flow analyses, the anticipated reduction in operating margin for Cycle 32 and later cycles, and the fact that the remainder of operating B&W plants do not have an equivalent TS condition that allows operating with only one RCP in operation per loop, Entergy has determined that a change to the TSs is required.

2.4 Description of the Proposed Change

Following Cycle 31, the COLR will no longer be updated to contain curves for one-RCP per loop operation since the desired margin to support such operation may not exist for fuel Cycles 32 and later. As such, for future cycles, the two-RCP curves will be removed from the COLR. This amendment request proposes removal of TS 3.4.4 "RCS Loops – Modes 1 and 2," Condition A, "One RCP not in operation in each loop", Condition B be relisted as Condition A and revised to state, "Requirements of LCO not met." The revised Required Action is to be in Mode 3 within 6 hours.

Since extended two-RCP operation would no longer be allowed by TS 3.4.4, Entergy performed an analysis to determine if the current setpoint value for 1-RCP/loop operation remained valid. The analysis determined that no change to the PPM trip setpoint of ≤ 55% with 1-RCP/loop in operation is required.

This change results in TS 3.4.4 being consistent with NUREG-1430 "Standard Technical Specifications (STS) – Babcock and Wilcox Plants," Revision 5 (Reference 1). NUREG-1430 Table 3.3.1-1 Function 7 has a bracketed default value of "[5]% RTP with ≤ 2 pumps operating." The PPM analysis demonstrates that there are no adverse effects due to leaving this value at the current TS Allowable Value of "≤ 55% RTP with one pump operating in each loop."

The TS bases for TS 3.4.4 and TS 3.3.1 are being revised to support the change mentioned above. In addition, the basis for TS 3.4.1 "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits" is being revised to remove the discussion concerning two-RCP operation. Refer to NUREG-1430 "Standard Technical Specifications (STS) – Babcock and Wilcox Plants Bases" (Reference 2).

3.0 TECHNICAL EVALUATION

The ANO-1 TSs place limits on allowable reactor RTP based on running RCP combinations. Full power operation is allowed if four RCPs are running based on inputs from the RCP status monitors. The allowable power level setpoint is adjusted automatically depending on the combination of running RCPs. If two RCPs are running (one on each loop), reactor power is limited to ≤ 55% RTP by the PPM trip in RPS. The reactor is tripped regardless of reactor power if there are no RCPs running on either RCS loop.

The power / imbalance / flow reactor trip also serves to protect the fuel from DNB. RCS flow is measured on each RCS hot leg, and an RCP trip, sheared shaft, or locked rotor will cause measured RCS flow to lower. As flow drops during RCP coastdown, the reactor power trip setpoint on the power / imbalance / flow modules lowers in response. The curves that determine the maximum power for a corresponding RCS flow (adjusted for imbalance) are provided in each fuel cycle's COLR. The separate PPM trip relies on inputs from the RCP monitors which detect a low current condition on each of the RCPs. This is an instantaneous trip based on the number of RCPs in operation in each RCS loop vs RTP.

3.1 SAR Chapter 14 Evaluations

Chapter 14 of the ANO-1 Safety Analysis Report (SAR) contains the accident analyses associated with DBAs and evaluation of other non-DBA events. The Loss of Coolant Flow (LOCF) accident (SAR Section 14.1.2.6) states that "… one and two pump coastdown events have also been addressed in the determination of limiting safety system settings for the RPS overpower trip based on flow and imbalance." The proposed change does not invalidate this statement.

ANO-1 SAR Section 14.1.2.6.1

> *The reactor coolant flow rate is reduced if one or more of the Reactor Coolant Pumps fail. A pumping failure can occur from mechanical failures or from a loss of electric power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others. The mechanical failure considered in the analyses is the locked pump rotor in which a pump stops instantaneously, with a very rapid reduction in flow. …*
>
> *Faults in an individual pump motor or its power supply could cause the loss of one pump with a corresponding reduction in flow as the pump coasted to a stop. A failure of a 6,900-volt bus could cause the loss of two pumps. A complete loss of forced flow is unlikely and would occur only if the primary source of offsite power were lost simultaneously with loss of the Unit Auxiliary Transformer (UAT). A complete power loss would cause immediate reactor trip independent of protection system actuation.*
>
> *The original ANO-1 accident analyses specifically addressed the four pump coastdown, demonstrating that core damage would not result from this low probability event. Although not part of the original accident analyses, one and two pump coastdown events have also been addressed in the determination of limiting safety system settings for the RPS overpower trip based on flow and imbalance.*

ANO-1 SAR Section 14.1.2.6.2

> *The criterion for reactor protection for the pump coastdown loss-of-flow event is that the minimum DNB ratio shall not be less than the applicable critical heat flux correlation limit [e.g., 1.3 for BAW-2 correlation (Reference 6), 1.18 for the BWC correlation (Reference 7), or 1.132 for the BHTP correlation (Reference 8)]. For the locked rotor accident, the minimum DNB ratio shall not be less than applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions.*

The PPM trip is used for transient and steady-state protection of the reactor. The loss of flow analyses in the SAR are subdivided into two events: 1) loss of all forced reactor coolant flow and 2) locked RCP rotor. The analysis methodology for the four RCP coastdown and locked rotor events is similar. The major differences between the events are the rate of core flow reduction and the RPS trip response used as analysis inputs. With the reactor at power, the result is an increase in the RCS temperature and a reduction in the heat removal capability of the reactor coolant. These two conditions could result in DNB in the core. Significant analysis input parameters used for the four RCP coastdown and locked rotor events are presented in SAR Table 14-11 "Loss of Coolant Flow Accident Parameters for Original Four Pump Coastdown Analysis" and Table 14-12 "Locked Rotor Accident Original Analysis Parameters." None of the parameters listed in the aforementioned tables and relevant analyses are adversely affected by the proposed change to remove the two-RCP curves from the COLR nor is the proposed change to TS 3.3.4 considered adverse resulting in a reduction in safety.

In addition to the loss of all forced reactor coolant flow accident and the locked rotor accident analyses, Section 14.1.2.6.3 of the SAR mentions the two-RCP coastdown loss of flow analysis, which is not considered part of the ANO-1 accident analysis. The two RCP coastdown analysis, which is performed as part of the core reload analyses, is used to determine the setpoint for the power-to-flow portion of the power / imbalance / flow RPS trip. The two RCP coastdown is more limiting than the four RCP coastdown because the RPS trip credited in the two RCP coastdown event (power to flow trip) is slower than the trip credited for the four RCP coastdown event (PPM trip). Methodology similar to that described in the SAR for the four pump coastdown is used for the two pump coastdown analyses.

With the transition of the B&W plants to the Standard Technical Specifications (References 1 and 2), the PPM trip became the primary trip and flux/flow the secondary trip. The methodology identified that the four-to-two RCP coastdown and three-to-one RCP coastdown events (LOCF transients evolving to two-RCP flow conditions) needed to be justified for the PPM (1/1) overpower portion of the setpoint. Both accident scenarios were already analyzed using the flux/flow setpoint, but it was also required to demonstrate that the PPM (1/1) trip provided protection for these events as well. The method developed was to determine a maximum allowable power for two-RCP flow conditions by running steady-state DNB cases at increasing power levels until the minimum core DNBR approached the applicable critical heat flux (CHF) correlation limit. The steady-state analysis approach was determined to be conservative since the flow during either LOCF event (four-to-two RCP coastdown or three-to-one RCP coastdown) is always greater than the steady-state two pump flow. In addition, the two points on the pressure-temperature (P-T) curve where the steady-state DNB analyses are performed are the most restrictive of any steady-state condition and bound all other two-RCP P-T combinations.

After determining the maximum allowable power, the PPM (1/1) TS setpoint was then established. Historical calculations for several B&W plants demonstrated that the PPM (1/1) TS

setpoint could be established at values greater than 65% full power (FP); however, the final PPM (1/1) Allowable Value (TS Table 3.3.1-1 Function 7) was ultimately set at a conservative value of ≤ 55% RTP at ANO-1 for the TS Table 3.3.1-1. This approach precluded the need to further demonstrate that the PPM (1/1) trip protects the four-to-two RCP coastdown and three-to-two RCP coastdown events on a plant- or cycle-specific basis. When considering a scenario where the plant may be operating with 2 RCPs during TS 3.4.4 LCO conditions, both the PPM (1/1) trip and the flux/flow trip provide steady-state DNB overpower protection.

Since the accident analyses do not model the PPM setpoint, the response characteristics are used in place of the accuracy requirements. The equipment accuracy is accounted for in the response time of the string. If the response time can be met, the instrumentation is deemed to be operating correctly. Therefore, the response time requirement is the only performance criterion placed on the RCP status string.

As modeled in the accident analyses, the PPM trip is a digital type trip. If the accident analysis assumes a loss of primary flow from a reactor coolant pump trip, the trip condition is based on the initial core power. If the core power is greater than the allowed final pump status power level, a trip condition exists, and a reactor trip is initiated. For example, a four RCP coastdown transient results in a final RCP configuration of 0/0, but the PPM will trip the plant for all idle loop conditions if the power level is greater than 0% FP. Consequently, the 4 RCP coastdown transient analysis models the PPM trip signal at time zero with the control rods beginning to drop after the PPM trip delay time.

ANO-1 has non-redundant power-pump monitors. A single failure of a monitor would affect all four RPS channels. The primary trip for the two RCP coastdown event from four RCP operation at 100% FP is an immediate trip on PPM of 0% RTP (0/2, 2/0) or on PPM of 55% RTP (1/1). However, in accident analysis space, the single failure requirement results in no PPM trip. Consequently, the four-to-two RCP coastdown event produces the lowest DNBR and sets the basis for the RPS flux/flow setpoint at ANO-1.

The PPM analysis summarizes the ANO-1 LOCF accidents that must be considered and the RPS setpoints that protect each scenario. The analysis shows that all LOCF events protected by the primary PPM trip will trip immediately regardless of whether the PPM trip is 0% RTP or 55% RTP since the initial power for four RCP or three-RCP normal operating conditions is greater than 55% RTP. For a scenario where the plant is operating at a power level less than 55% FP, a four-to-two RCP coastdown or three-to-two RCP coastdown event would not initiate a plant trip. This would allow the operator to begin a controlled cooldown to Mode 3 to be in compliance with LCO requirements outlined in the proposed change to TS 3.4.4.

As previously described, the PPM trip does not place any requirement on the allowed power level for four- or three-RCP operation, which allows the RPS high flux and flux/flow trips to maintain the required protection. For a scenario where two-RCP operation may occur for a short period of time, as defined by TS 3.4.4 LCO, both the RPS flux/flow and PPM (1/1) setpoints provide overpower protection. Normal two-RCP operation is not permitted outside of the LCO condition since the required transient analyses have not been performed to support ANO-1 SAR Chapter 14 accidents initiated from two-RCP operating conditions.

The results of the PPM analysis state that 62.833% FP is the maximum real power that could be achieved under two-RCP (1/1) flow conditions if the RPS PPM (1/1) trip setpoint of 55% RTP is

reached. Also, the ANO-1 PPM (1/1) trip setpoint could be as high as 62.3% RTP and still provide the required steady-state and transient protection.

Summary

Entergy has reviewed the RCS and RPS licensing bases and determined that none are adversely impacted by the proposed change. Based on adopting a more restrictive TS Condition in TS 3.4.4, Entergy concludes that the proposed change meets applicable requirements and does not adversely affect nuclear safety.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Entergy proposes to remove the allowance in TS 3.4.4 to operate with only two RCPs in operation (one per RCS loop) for up to 18 hours before entering a shutdown timeclock to ensure adequate operating margin exists for Cycle 32 and future cycles.

ANO-1 was not licensed to the 10 CFR 50, Appendix A, GDC. ANO-1 was originally designed to comply with the 70 "Proposed General Design Criteria for Nuclear Power Plant Construction Permits," published in July 1967. However, the ANO-1 SAR provides a comparison with the Atomic Energy Commission (AEC) GDC published as Appendix A to 10 CFR 50 in 1971. The applicable AEC GDC were compared to the 10 CFR 50, Appendix A, GDC as discussed in Section 1.4 of the ANO-1 SAR.

Applicable AEC GDCs associated with RCS or RPS that may be affected by the proposed change are listed below:

GDC 10, "Reactor Design"

GDC 13, "Instrumentation and Control"

GDC 15, "RCS Design"

GDC 20, "Protection System Functions"

ANO-1 Design/Licensing Bases: Reactor Coolant System (RCS)

GDC 10 "Reactor Design": The ANO-1 reactor meets the regulatory requirements specified in GDC 10 of 10 CFR Part 50, Appendix A. It requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. It requires that the reactor core and the RCS be designed with the necessary margins to accommodate, without fuel damage, transients that are expected to occur one or more times during the life of the plant.

With respect to meeting the intent of GDC 10 and reactor design, this SAR section states:

> *The reactor is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady state operation, including the transients listed in the criterion. The integrity of fuel cladding is assured under all normal and abnormal modes of anticipated operation by avoiding overstressing and overheating of cladding. The core design, together with reliable process and Decay Heat Removal Systems, provides for this capability under all expected operating conditions and transients.*
>
> *The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine mismatch. Above approximately 22 percent rated power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to dampen the effects of power transients. The RCS will maintain the reactor operating parameters within preset limits and the RPS will shut down the reactor if normal operating limits are exceeded by preset amounts.*
>
> *The RCPs have sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated.*

The proposed change does not alter the design of the reactor core and associated coolant, control, and protection systems functions. It removes the allowance to operate with only one RCP per loop in operation for more than 6 hours before entering Mode 3. This change is more restrictive and conservative than the current TS and does not reduce any design margins.

GDC 15 "RCS Design": The ANO-1 RCS meets the regulatory requirements specified in GDC 15 of 10 CFR Part 50, Appendix A. It requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

With respect to meeting the intent of GDC 15 and RCS design, SAR section 1.4 states:

> *An analysis and evaluation of all normal and abnormal operating conditions and transients is integrally related to all RCS and associated systems design. For all anticipated transients, plots of critical variables, e.g. temperature, pressure, and the rate of temperature change, are generated for critical components. Also, the number of lifetime cycles was determined for each transient. All of these analysis results were invoked as functional requirements on the detailed design of the affected systems. Margins for uncertainties were included in (1) the basic analysis assumptions, (2) the assessment of lifetime cycles, and (3) in the code dictated procedures for stress analysis.*

The proposed change does not alter the design of the control and protection instrumentation that is provided to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

ANO-1 Design/Licensing Bases: Reactor Protection System (RPS)

GDC 13 "Instrumentation and Control": The ANO-1 RPS meets the regulatory requirements specified in GDC 13 of 10 CFR 50, Appendix A. It requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

With respect to meeting the intent of GDC 13 and RPS design, SAR section 1.4 states:

> *Adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety.*
>
> *Instrumentation systems include the nuclear instrumentation system, which monitors the neutron flux level from source to 125 percent of rated power; the non-nuclear process instrumentation which measures temperatures, pressures, flows, and levels in the RCS, steam system, and reactor auxiliary systems; and the incore instrumentation system which measures neutron flux at specific locations within the reactor core.*
>
> *Control is provided by two basic systems, the first of which is:*
>
> > *A. Protection Systems*
> >
> > *The protection systems, which consist of the Reactor Protection System and the safeguards actuation system, perform the most important control and safety functions. The protection systems extend from the sensing instruments to the final actuating devices, such as circuit breakers and pump or valve motor contactors. …*

The proposed change does not alter the design of the instrumentation that is provided to monitor variables and systems over their anticipated ranges for normal operating, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety.

GDC 20 "Protection System Functions": The ANO-1 RPS meets the regulatory requirements specified in GDC 20 of 10 CFR 50, Appendix A. It requires that the protection system be designed (1) to automatically initiate the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

With respect to meeting the intent of GDC 20 and RPS design, SAR section 1.4 states:

> *Safety analyses have been conducted to assure that acceptable fuel design limits are not exceeded during operational occurrences. The Reactor Protection System design meets this criterion by monitoring variables that sense the accident condition and provide a reactor trip and safety features action. The Reactor Protection System limits*

> *reactor power that might result from unexpected reactivity changes and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits. …*

The proposed change does not alter the design of reactivity control protection systems or instrumentation that sense accident conditions to initiate systems or components important to safety.

NUREG-0800, Revision 2, "Standard Review Plan," Section 15.3.1, "Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions," (Reference 9) indicates that the licensee's flow transient analyses are reviewed to ensure that values of pertinent system parameters remain within expected ranges. A complete or partial loss of forced reactor coolant flow should not result in exceeding reactor parameters.

Effects of the Proposed Change on Ability of ANO-1 to Meet GDC 10/13/15/20

The proposed change to the ANO-1 TSs and COLR does not adversely affect ANO-1's ability to meet GDC 10, 13, 15, and 20. Removing TS 3.4.4 Condition A prevents extended power operation with one RCP in operation in each loop which enhances plant safety by preventing operation in conditions where the reactor and RCS safety margins are not fully analyzed for all conditions.

The proposed change was compared with other B&W sites' (Davis Besse and Oconee) TSs. Neither site allowed for indefinite two-RCP operations in TS 3.4.4. Davis Besse's PPM trip setpoint is ≤ 55.1% RTP and Oconee's setpoint is ≤ 2% RTP for one RCP in operation per loop. The STS (Reference 1) TS 3.3.1 Table 3.3.1-1 Function 7 value (bracketed) is ≤ 5% RTP for one RCP in operation per loop. The basis for this value is to prevent normal power operation unless at least three RCPs are operating (Reference 2). The PPM analysis demonstrates that leaving the current RTM trip setpoint at ≤55% RTP with one RCP in operation per loop is acceptable. As such, the proposed change continues to ensure that the DNBR and fuel centerline melt SLs are not exceeded.

10 CFR 50.36 Technical Specifications Criteria

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses include TS as part of the facility operating license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. Pursuant to 10 CFR 50.36, TS are required to include items in the five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

However, the rule does not specify the particular requirements to be included in a plant's TS. Under 10 CFR 50.36(c)(2)(ii), a limiting condition for operation must be included in TS for any item meeting one of the following four criteria:

> Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The proposed change maintains the applicable requirements for RCS loops (i.e., the required SSCs) in the TS in accordance with Criterion 3 of the regulation. The RCS loop requirements and RPS power-to-pumps setpoint Allowable Value continue to be specified in the TS LCOs and continue to be verified by the TS SRs as required by the regulation.

4.2 Precedent

No relevant precedent license amendments could be found where TS 3.4.4 was changed to eliminate two-RCP operation with one RCP operating per loop. Only three US sites continue to operate B&W reactors: ANO-1, Davis Besse, and Oconee. A review of the Davis Besse and Oconee TS for RCS Loops revealed that neither site permits normal operation with only one RCP operating per RCS loop. Their TS are similar to the STS and require the plants to enter Mode 3 (Davis Besse within 6 hours, Oconee within 12 hours) if only one RCP is operating on each RCS loop.

Davis Besse's TS Table 3.3.1-1 for RPS Instrumentation has an Allowable Value of ≤ 55.1% RTP with one RCP operating in each RCS loop. Oconee's TS Table 3.3.1-1 sets the power-to-pumps trip to "> 2% RTP with ≤ 2 pumps operating." The basis for Oconee's lower limit (2% RTP) is "… to prevent normal power operation unless at least three RCPs are operating."

4.3 No Significant Hazards Consideration Analysis

Entergy Operations, Inc. (Entergy) requests a revision to Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS) to eliminate TS 3.4.4 Condition A which permits operation with one Reactor Coolant Pump (RCP) per loop for 18 hours without entering a TS required shutdown to Mode 3. The proposed change eliminates the concern of inadequate operating margin for Cycle 32 and beyond.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

The proposed change prohibits continued power operation for more than 6 hours with one RCP operating per loop by eliminating Condition A from the current TS 3.4.4. The change is necessary due to forecasted loss of operating margin for future operating cycles.

The accident related to the proposed change is the Loss of Coolant Flow Accident (SAR Section 14.1.2.6). This includes a reactor coolant pump (RCP) locked rotor, (RCP sheared shaft and RCP trip are bounded by the nearly instantaneous locked rotor event), and loss of power to one or more RCPs. The criterion for reactor protection for the pump coastdown loss-of-flow event is that the minimum departure from nucleate boiling (DNB) ratio shall not be less than the applicable critical heat flux correlation limit (e.g., 1.3 for BAW-2 correlation, 1.18 for the BWC correlation, or 1.132 for the BHTP correlation). For the locked rotor accident, the minimum DNB ratio shall not be less than applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions. The original Arkansas Nuclear One, Unit 1 (ANO-1) accident analyses specifically addressed the four pump coastdown, demonstrating that core damage would not result from this low probability event.

No new failure modes are introduced by the proposed technical specification (TS) change. The proposed change complies with industry and regulatory standards and acts to preserve safety margins. The change continues to provide the minimum DNB ratio for all evaluated accidents. The proposed change does not increase the potential for a loss of coolant flow accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change is more conservative than the current TSs. The current TSs allow extended operation of two RCPs (one RCP per loop) for 18 hours before entering a 6 hour required shutdown to Mode 3. However, the proposed change does not allow operation with one RCP per loop without entering a plant shutdown requiring the plant to be in Mode 3 within 6 hours. This change is consistent with the Standard Technical Specifications (Reference 1). No credible new failure mechanisms, malfunctions, or accident initiators that have not been previously considered in the design and licensing bases are introduced. The current accident analyses remain bounding with consideration of the proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

The proposed change prohibits continued power operation for more than 6 hours with one RCP operating per loop by eliminating Condition A from the current TS 3.4.4.

The proposed change assures sufficient safety margins are maintained, and that the design, operation, surveillance methods, and acceptance criteria specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant's licensing basis. The proposed change does not adversely affect the reliability of the equipment assumed to operate in the accident analysis or existing plant safety margins. Applicable accident analyses in which this change is applicable have been verified to remain bounding. As such, there are no changes being made to safety limits or limiting safety system settings that would adversely affect plant safety as a result of the proposed change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

4.4 Conclusion

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

5.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required for the proposed change.

6.0 REFERENCES

1. NUREG-1430, "Standard Technical Specifications – Babcock and Wilcox Plants," Revision 5, Volume 1, (ADAMS Accession No. ML21272A363), September 2021

2. NUREG-1430, "Standard Technical Specifications – Babcock and Wilcox Plants: Bases," Revision 5, Volume 2, (ADAMS Accession No. ML21272A370), September 2021

3 Babcock and Wilcox - BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Revision 9, (ADAMS Accession No. ML16106A285), November 2017

4 Babcock and Wilcox - BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Revision 3, (ADAMS Accession No. ML993080291), October 1999

5. Babcock and Wilcox - BAW-10103A, "Emergency Core Cooling System (ECCS) Analysis of Babcock and Wilcox's 177-Fuel Assembly (FA) Lowered Loop Nuclear Steam Supply (NSS)," Revision 3, July 1977

6. Babcock and Wilcox Topical Report BAW-10000(A), "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," (ADAMS Legacy Accession No. 8201060344), Lynchburg, VA, May 1976

7. Babcock and Wilcox Topical Report BAW-10143P-A, "BWC Correlation of Critical Heat Flux," (ADAMS Accession No. ML20195C088), dated April 1985

8. Framatome-ANP Topical Report BAW-10241P-A, "BHTP DNB Correlation Applied with LYNXT," (ADAMS Accession No. ML21015A129), dated February 2, 2021

9. NRC NUREG-0800, "Standard Review Plan," Revision 2, (ADAMS Accession No. ML070550010), dated March 2007

10. Framatome Condition Report CR-2021-0496 Evaluation, "Review of the Method for obtaining the ANO-1 two-RCP (one RCP in each loop) Setpoints at Overpower," dated April 2, 2021

11. Nuclear Energy Institute (NEI), NEI 15-03, Revision 3, “Licensee Actions to Address Nonconservative Technical Specifications,” (ADAMS Accession No. ML20100G899), dated March 2020

ATTACHMENTS

1. Technical Specification Page Markup

2. Technical Specification Bases Page Markups (Information Only)

3. Retyped Technical Specification Pages

Enclosure Attachment 1

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Technical Specification Page Markup
(1 page)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and THERMAL POWER restricted as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One RCP not in operation in each loop.~~	~~A.1 Restore one non-operating RCP to operation.~~	~~18 hours~~
~~B~~A. ~~Required Action and associated Completion Time of Condition A not met.~~ ~~OR~~ Requirements of LCO not met ~~for reasons other than Condition A~~.	~~B~~A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program

Enclosure Attachment 2

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Technical Specification Bases Page Markups (Information Only)
(7 pages)

APPLICABLE SAFETY ANALYSES (continued)

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing may be insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline temperature SLs.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides protection for the loss of a pump or pumps which would result in both pumps in a single steam generator loop being tripped.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least one RCP is operating in each loop. RCP status is monitored by power transducers associated with each pump. These relays indicate a loss of an RCP on underpower. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value -account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment. Analysis shows that the allowable value of 55% with one RCP in service per loop is adequate to prevent DNB in the event of an RCP trip.

The power to pump trip does not place any requirement on the allowed power level for 4- or 3-RCP operation, which allows the RPS high flux and flux/flow trips to maintain the required protection. For a scenario where two-RCP operation may occur for a short period of time, as defined by TS 3.4.4 LCO, both the RPS flux/flow and power to pumps (one pump per loop) setpoints provide overpower protection. Normal two-RCP operation is not permitted outside of the LCO condition since the required transient analyses have

not been performed to support ANO-1 SAR Chapter 14 accidents initiated from two-RCP operating conditions.

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the reactor core SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

REFERENCES

1. SAR, Chapter 7.

2. SAR, Chapter 14 and Chapter 3A.

3. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.

4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

5. 10 CFR 50.36.

6. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985.

7. CALC-22-E-0001-11, "ANO-1 TS 3.4.4 LAR Support – RPS Pump-to-Power Monitor Setpoint."

APPLICABLE SAFETY ANALYSES (continued)

The safety analyses to establish reload operating limits are performed using nominal values for RCS coolant average temperature, core outlet pressure, and RCS flow rate and core power level with appropriate application of associated uncertainty. Consistent with Statistical Core Design (SCD) methodology, applicable random parametric uncertainties are combined statistically. As necessary, bias parameters are included deterministically. The RCS temperature and pressure are measured in the hot leg. The surveillance criteria specified in the COLR include adjustment for measurement location. The COLR specified hot leg temperature is the maximum allowed so that the analysis value is not exceeded. The COLR specified hot leg pressure and flow are the minimum allowed so that the analysis values are not exceeded.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for ~~two pump,~~ three pump and four pump operation. The flow limits for ~~two pump and~~ three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops-MODES 1 and 2").

The steady state limits on DNBR related parameters are provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive on plant operations than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, a check must be performed to determine whether an SL may have been exceeded.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure and temperature limits are to be applied to the loop with two reactor coolant pumps (RCPs) running for the three RCPs operating condition.

The surveillance criteria for pressure, temperature, and flow rate as specified in the COLR have been appropriately adjusted for the measurement location and for instrument error consistent with supporting analysis.

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS hot leg temperature, and RCS flow rate must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a significant concern.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 1) contain various assumptions for the Design Bases Accident (DBA) initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of RCS flow on DNB. The initial condition DNB protection for the limiting loss of coolant flow event for four and three pump operation is provided by the RCS flow surveillance criteria specified in the COLR for SR 3.4.1.3 and SR 3.4.1.4. The loss of coolant flow event which has been found to produce the limiting DNB is the four-to-two pump coastdown. In addition to the coastdown events, the single pump locked rotor event has been analyzed and shows that either the minimum DNB ratio is not less than the applicable critical heat flux correlation limit, or fuel cladding was shown to experience no significant temperature excursions.

Steady state DNB analysis has been performed for four and three pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature SL (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 112% RTP. This is the design overpower condition for four pump operation. The 112% value is the accident analysis limit of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR that protects the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the RPS nuclear overpower RCS flow and measured AXIAL POWER IMBALANCE Function. The maximum power level for three pump operation is identified in the COLR and is based on the three pump flow as a fraction of the four pump flow at full power.

Although the Specification limits operation to a minimum of three pumps total, design evaluation (including analyses at steady state, ECCS initial conditions, and DNB conditions) also shows that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is ~~restricted to 24 hours~~not allowed by this specification. ~~(Ref. 2) since not all transient and accident conditions have been analyzed.~~

RCS Loops-MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal via two RCS loops. An operating loop consists of at least one operating RCP and a SG capable of heat removal. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if fewer pumps are available, power must be reduced as specified in the COLR.

APPLICABILITY

In MODES 1 and 2, the reactor may be critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

~~A.1~~

~~With one RCP not in operation in each loop, the assumptions of the safety analyses are not met, but design evaluation provided in Reference 2 concludes that events initiated during two pump operation would be expected to respond within the acceptance criteria for the ECCS. However, since no analysis was performed, Technical Specifications for two pump operation will only allow operation in MODES 1 or 2 for a period not to exceed 24 hours. The Completion Time of 18 hours provides sufficient time to restore operation of an additional RCP, while allowing time to place the unit in MODE 3 within the 24-hour limitation if restoration of a third RCP is not accomplished.~~

~~B~~A.1

If the ~~Required Action and associated Completion Time of Condition A~~requirements of the LCO are not met, ~~or if the LCO is not met for any reason other than provided in Condition A,~~ the unit must be placed in a MODE in which the requirements are not applicable. This is accomplished by placing the unit in MODE 3. This reduces the core heat removal needs and minimizes the possibility of violating DNB limits. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

1. SAR, Chapters 14 and 3A.

~~2. BAW-10103A, Revision 3, July 1977.~~

~~3~~2. 10 CFR 50.36.

Enclosure Attachment 3

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Retyped Technical Specification Page
(1 page)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and THERMAL POWER restricted as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program