Florida Power & Light Company, 9760 S.W. 344 St. Homestead, FL 33035

FPL

OCT 1 5 2011
L-2011-438
10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.

(2) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information – Round 1.3 (Part 3)", Accession No. ML11202A174, July 21, 2011.

(3) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-233), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11221A227, August 5, 2011.

(4) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 2.3 (Part 3)," Accession No. ML11263A204, September 20, 2011.

(5) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-400), "Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205," Accession No. ML11276A080, September 30, 2011.

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point (PTN) Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an Extended Power Uprate (EPU).

By email dated July 21, 2011[Reference 2], the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM) issued a Reactor Systems Branch (SRXB) Request for Additional Information (RAI) to FPL consisting of thirty-nine (39) questions on the loss of coolant accident (LOCA) and non-LOCA safety analyses presented in Reference 1. On August 5, 2011, FPL provided its response to RAI questions SRXB-1.3.1-1.3.6 and SRXB-1.3.16-1.3.39 via FPL letter L-2011-233 [Reference 3].

By email from the NRC PM dated September 20, 2011 [Reference 4], FPL received two follow-up RAIs requesting PTN to demonstrate acceptable plant response for two events that had not been previously analyzed: Inadvertent Opening of a Power-Operated Relief Valve (PORV) and Feedwater Line Break (FWLB) events. The results for the Inadvertent Opening of a PORV

A001
NRR

event (RAI SRXB-2.3.1) were issued to the NRC via letter L-2011-400 dated September 30, 2011 [Reference 5]. The results for the FWLB event (RAI SRXB-2.3.2) are presented in Attachment 1 to this letter.

RAI SRXB-1.3.21 of Reference 2 requested FPL to demonstrate that sufficient response time is available for operators to terminate a Boron Dilution event occurring in Modes 3, 4, or 5. The response to SRXB-1.3.21 maintained that a Boron Dilution event in Modes 3, 4, or 5 was beyond the current licensing basis for Turkey Point. During a follow-up telephone discussion between FPL and the NRC held on September 21, 2011, the SRXB staff requested that FPL supplement the response to SRXB-1.3.21 with specific analysis results generated for EPU conditions. The results of the requested analyses for Boron Dilution events in Modes 3, 4, and 5 are presented in Attachment 2 to this letter.

In support of the new Boron Dilution analysis, Turkey Point is increasing the minimum required shutdown margin maintained in Mode 4 (without Reactor Coolant Pumps operating) and in Mode 5. Proposed amendments to the Turkey Point Technical Specifications reflecting these changes are included with the Boron Dilution analysis results in Attachment 2.

The Turkey Point Plant Nuclear Safety Committee (PNSC) has reviewed the proposed amendments.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c). FPL has determined that the proposed changes do not involve a significant hazards consideration. Also, the proposed changes do not alter the environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 15, 2011.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments (2)

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4

RESPONSE TO NRC REACTOR SYSTEMS BRANCH REQUEST FOR ADDITIONAL INFORMATION REGARDING EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST NO. 205

ATTACHMENT 1

SRXB-2.3.2

Response to Request for Additional Information – Feedwater Line Break Analysis

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

By email dated July 21, 2011[Reference 2], the NRC Project Manager (PM) issued a Reactor Systems Branch (SRXB) RAI to FPL consisting of thirty-nine (39) questions on the loss of coolant accident (LOCA) and non-LOCA safety analyses presented in Reference 1. On August 5, 2011, FPL provided its response to RAIs SRXB-1.3.1-1.3.6 and SRXB-1.3.16-1.3.39 via FPL letter L-2011-233 [Reference 3].

By email from the NRC PM dated September 20, 2011 [Reference 4], FPL received two follow-up RAIs requesting PTN to demonstrate acceptable plant response for two events that had not been previously analyzed: Inadvertent Opening of a Power-Operated Relief Valve (PORV) and Feedwater Line Break (FWLB) events. The results for the Inadvertent Opening of a PORV event (RAI SRXB-2.3.1) were issued to the NRC via letter L-2011-400 dated September 30, 2011 [Reference 5]. The results for the FWLB event (RAI SRXB-2.3.2) are presented below.

SRXB-2.3.2 Licensing Report §2.8.5.2.4, "Feedwater System Pipe Breaks Inside and Outside Containment", states that feedwater system pipe breaks are not required to be analyzed per the PTN current licensing basis. Generally, the feedwater line break (FWLB) analysis results are used to:

(1) determine the performance requirements of the auxiliary feedwater system, as a post-trip decay heat removal system, under the conditions/configuration that would exist following a FLWB,

(2) set the low steam generator water level reactor trip setpoint,

(3) determine the error allowance, on the low steam generator water level reactor trip setpoint, needed to compensate for the hostile environment that could be created by the break flow from a FWLB occurring inside containment,

(4) verify the capability to establish and maintain natural circulation cooling by showing that saturation conditions are not created inside the reactor coolant system, by a FWLB, and

(5) show that the core remains covered and that the integrity of the reactor coolant system pressure boundary is maintained during a FWLB (i.e., when there is water relief though the pressurizer safety valves).

For the Turkey Point units, provide this information, from the results of a FLWB analysis, or from another, equivalent source/method.

To address the above RAI, a representative evaluation has been performed using the standard Westinghouse safety analysis methodology to analyze a single FWLB case for auxiliary feedwater (AFW) capacity and margin to hot leg saturation. The case under consideration takes conservative input assumptions that normally yield the most limiting results for plants of similar design to PTN.

The current licensing basis for PTN does not include an analysis for the FWLB event, so the evaluation and results presented below are to be distinguished from those of a typical design basis safety analysis. The evaluation demonstrates that a FWLB event, with input assumptions typical of the limiting analyzed case, affords adequate margin to hot leg saturation under EPU conditions. This information provides reasonable assurance that the consequences of a FWLB event do not present a safety concern for operation at EPU conditions. As such, there is no need to incorporate a design basis safety analysis into the PTN licensing basis for EPU.

Key analysis inputs for this representative case are consistent with those that are typically limiting for Westinghouse plants with feedring-type steam generators (SG). These inputs include modeling the largest double-ended break possible for the Turkey Point SG model and minimum reactivity feedback parameters. A summary of the analysis is presented below.

<u>Event Description:</u>

The FWLB incident is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the SGs. If the break is postulated in a feedline between the check valve and the SG, fluid from the SG will be discharged through the break. Furthermore, with feedring-type SGs, because the AFW piping connects to the main feedline, a break between the check valve and the SG could preclude the subsequent addition of AFW to the affected SG. In contrast, if a break occurs upstream of the feedline check valve, the transient progresses as a Loss of Normal Feedwater event. Depending upon the size of the break and the plant operating conditions at the time of the rupture, the break could cause either a cooldown (by excessive energy discharge through the break) or heatup of the reactor coolant system (RCS). Because the consequences of an RCS cooldown resulting from a FWLB are bounded by the cooldown consequences of a Steam System Piping Failure, the FWLB event is analyzed only with respect to RCS heatup effects.

As the subcooled feedwater flow to the SGs is reduced by a FWLB, the long-term capacity of the secondary system to remove heat from the RCS is diminished. The feedwater flow reduction can cause RCS temperatures to increase prior to reactor trip. Additionally, fluid inventory of the faulted SG may be discharged through the break, which will reduce the heat sink volume available for decay heat removal following a reactor trip. The FWLB event is analyzed to demonstrate the ability of the AFW system to adequately remove long-term decay heat and prevent excessive heatup of the RCS.

In the analysis performed, the break is assumed to be located in a feedline between the check valve and the SG. A break in this location results in the discharge of fluid from the associated SG. The size of the break and the functionality of the main feedwater (MFW) control system are two important factors during a FWLB transient. Some breaks may be small enough such that a properly functioning MFW control system will be able to completely make up for the resultant inventory loss. In contrast, larger feedline breaks can cause a sizeable blowdown (inventory loss) that prevents the MFW control system from being able to supply enough feedwater to maintain shell-side fluid inventory in the SGs. This then leads to a low-low SG water level reactor trip and AFW actuation. Another important factor during a FWLB transient is the shell-side fluid inventory in the unaffected/intact SGs at the time of reactor trip. It is conservative to minimize this fluid inventory because it minimizes the heat removal capacity of the SGs, which maximizes the RCS heatup. For this purpose, the analysis performed assumes that MFW is completely terminated at the time the break occurs. Furthermore, the water level in the faulted SG is assumed at its highest level consistent with full-power conditions (to delay reactor trip), while the water level in each of the intact SGs is at its lowest (minimizes inventory for long-term heat removal).

Following the break, there is a rapid decrease in the SG inventory, a fast increase in the average reactor coolant temperature, a surge of water into the pressurizer with a resultant pressure increase in the RCS, as well as a pressure increase in the Main Steam System (MSS). When the SG water level reaches the low-low reactor protection setpoint, a reactor trip occurs, and AFW is initiated. A turbine trip is modeled to occur shortly after reactor trip to further reduce the heat removal capability of the SGs. No credit is taken for the secondary-side, non-safety-related power-operated relief valves or atmospheric steam dump valves. The turbine trip causes a sudden reduction in steam flow and a further reduction in the heat removal capacity of the SG. With the reduced steam flow, the SG pressure in each of the intact loops rapidly increases to the setpoint of the first (lowest setpoint) SG safety valves, and remains there until the RCS heatup ceases, i.e., until the heat removal capability of the SGs being fed AFW is sufficient to remove the decay heat generated in the core (also known as the time of event turnaround). During the heatup period after reactor trip, the pressurizer pressure increases to, and is maintained near, the pressurizer PORV setpoint. At event turnaround, the RCS temperature and pressure, and the pressurizer water level begin to decrease again, and the transient, as analyzed, is over. Subsequently, the plant operators can follow the applicable emergency operating procedures to first bring the plant to a stabilized temperature condition using the pressurizer and SG power-operated relief valves (if offsite power is available), and then eventually to a cold shutdown condition.

Unless the effects of the FWLB and subsequent SG water level reduction are counteracted by manual or automatic action, the rise in RCS temperature could eventually result in a loss of sub-cooled margin in the RCS hot or cold legs, and/or a challenge to the integrity of the RCS and MSS pressure boundaries. However, with respect to peak RCS and MSS pressures, the FWLB event is bounded by the Loss of Load/Turbine Trip event, in which assumptions are made to conservatively calculate the RCS and MSS pressure transients. For the FWLB event, turbine trip occurs after reactor trip, whereas for the Loss of Load/Turbine Trip event the turbine trip is the initiating fault. Therefore, the primary-to-secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are always more severe for Loss of Load/Turbine Trip than for FWLB. For this reason, the analysis performed makes no attempt to calculate the maximum RCS or MSS pressures for the FWLB event.

Instead, the intent of the analysis is to maximize the potential for reaching saturated conditions in the RCS hot and/or cold legs. Some of the key characteristics of the analysis are described below.

Method of Analysis:

The FWLB transient is analyzed by employing the detailed digital computer code RETRAN. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Normal reactor control systems are not required to function. The rod control system is in the manual mode, which allows for a conservative increase in power and average coolant temperature until reactor trip occurs. The reactor protection system functions to trip the reactor on the appropriate signal, while the Engineered Safeguards Features Actuation System (ESFAS), primarily the AFW system, is actuated to provide long-term decay heat removal capability. No single active failure will prevent the reactor protection system from functioning properly. The appropriate single active failure in the AFW system has been accounted for to minimize the available AFW flow to the intact SGs.

Inputs were modeled to maximize the potential for reaching saturated conditions in the hot and/or cold legs, as provided in Table SRXB-2.3.2-1 below.

The analysis of this event is performed with the reactor initially operating at full power conditions. This is bounding because the stored energy of the primary system is maximized at full power conditions, and therefore the disparity between the heat generated in the RCS and the heat removal capacity of the SGs is maximized.

The case analyzed corresponds to the largest possible break size. A dual-unit loss of offsite power following reactor trip is also considered as this results in additional challenges to the shared AFW system (i.e., lower AFW flows are available to the faulted unit).

Reactor trip is assumed to occur on a SG low-low water level signal received when level reaches 0% of the narrow range span (NRS). Should this reactor trip signal not be generated due to a potentially harsh environment caused by the FWLB inside containment, it is estimated that a high containment pressure signal would trigger a safety injection signal and subsequent reactor trip a few seconds after the SG low-low level setpoint is reached, and in sufficient time to ensure that the acceptance criteria are met for the event.

One of the three available AFW pumps is capable of supplying a minimum flow of 96 gpm to each of the two intact SGs following a 95-second delay and for up to 10 minutes following receipt of the AFW initiation signal; at 10 minutes, the AFW flow is increased to 125 gpm to each intact SG following appropriate operator actions to terminate flow to the faulted SG. The AFW flow is initiated 95 seconds after the low-low SG water level setpoint is reached. The AFW flow conditions were based on a maximum temperature of 106°F.

Results:

A time sequence of events is presented below in Table SRXB-2.3.2-2. Additionally, the transient plots for this analysis are provided in Figures SRXB-2.3.2-1 through SRXB-2.3.2-9 below. Note that a steady-state run of 100 seconds precedes the FWLB transient.

From the results obtained, the available AFW capacity was determined to be adequate for long-term decay heat removal and prevents saturated conditions in the hot and cold legs from being reached during this transient (which conservatively demonstrates that the core remains covered). The results presented below are based on a conservatively low SG low-low level setpoint of 0% NRS. However, should the harsh environment resulting from the break prevent this reactor trip function from generating a signal, the amount of water vapor in containment is expected to be such that a back-up function (specifically, a containment high pressure signal) would be actuated in time to trip the reactor and still ensure that the applicable acceptance criteria for this event are met. Through a simplified calculation of the expected containment response following a double-ended FWLB, it was estimated that a containment high pressure signal would be reached roughly 5 seconds after the SG low-low level setpoint of 0% NRS is reached, or just over 10 seconds into the transient. Even with a delayed reactor trip derived from this containment signal, saturated conditions in neither the cold nor hot legs were reached.

Conclusion:

The results of the FWLB analysis confirm that the available AFW capacity is adequate to ensure that the applicable acceptance criteria for this event are met.

Table SRXB-2.3.2-1 – Critical Parameters for the Turkey Point Units 3 and 4 FWLB Analysis

Parameter	Value	Conservative Direction
Nuclear Steam Supply System (NSSS) Power Rating (MWt)	2652 + 0.3 % uncertainty(1)	High
Total RCS Flow Rate (gpm)	260700	Low
Moderator Temperature Coefficient (pcm/ °F)	0.0	High(2)
Doppler Power Coefficient (pcm/% power vs. power)	-9.55+0.037Q(3)	High(2)
Decay Heat	ANS 1979 + 2σ	High
Initial Pressurizer Pressure (psia)	2197	Low
Initial Pressurizer Water Level (% span)	67.1	High
Initial RCS Average Temperature (°F)	589.0	High
Initial SG Water Level (% NRS) – Faulted SG	56	High
Initial SG Water Level (% NRS) – Intact SGs	38	Low
SG Tube Plugging (%)	10.0	High
Low-Low SG Water Level Safety Analysis Reactor Trip Setpoint (% NRS)	0%	Low
Reactor Trip Delay Time on Low-Low SG Water Level (sec)	2	High
Delay for Initiation of AFW (sec)	95	High
AFW to Faulted SG (gpm)	0	Low
AFW to Each Intact SG (gpm)	96 up to 10 minutes, 125 after 10 minutes	Low
AFW Temperature (°F)	106	High
(1) 8 MWt of nominal pump heat was modeled up to the time the loss of offsite power occurs (2) Least Negative (3) Q = Percent of Nominal Power		

Table SRXB-2.3.2-2 Time Sequence of Events for the Turkey Point Units 3 and 4 FWLB Analysis

Event	Time (sec)
FWLB Transient Begins	100.0
Low-Low SG Water Level Reactor Trip Setpoint is Reached	105.4
Rod Motion Begins	107.4
Turbine Trip Occurs	107.9
Reactor Coolant Pumps Trip	109.4
AFW Starts	200.4
Final RCS Cooldown Begins	~3625

Figure SRXB-2.3.2-1 – Nuclear Power vs. Time

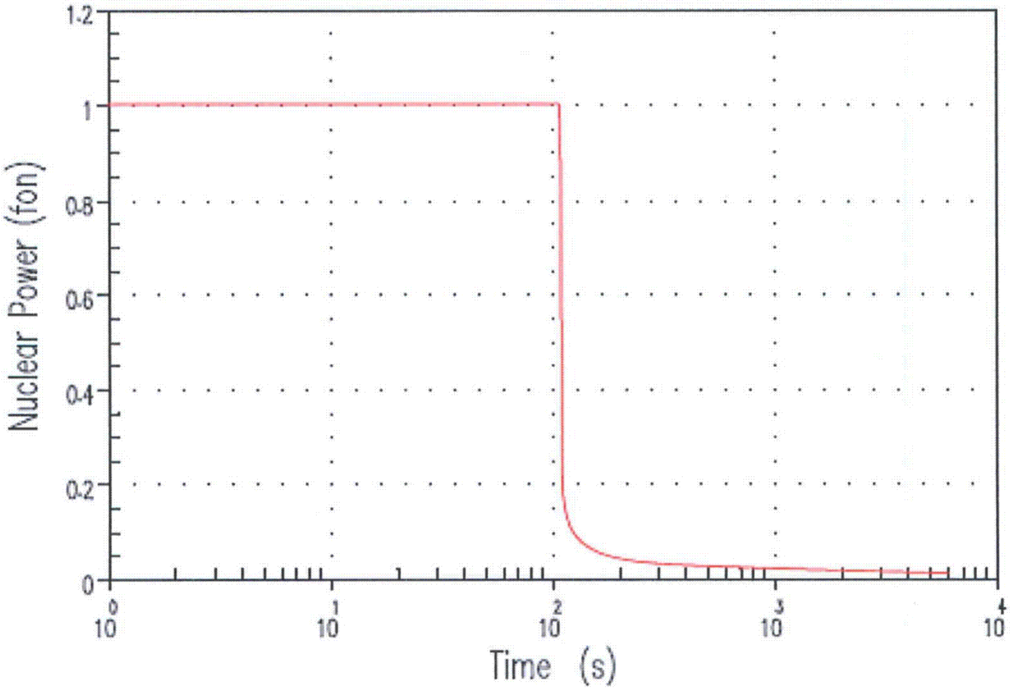


Figure SRXB-2.3.2-2 – Pressurizer Mixture Level vs. Time

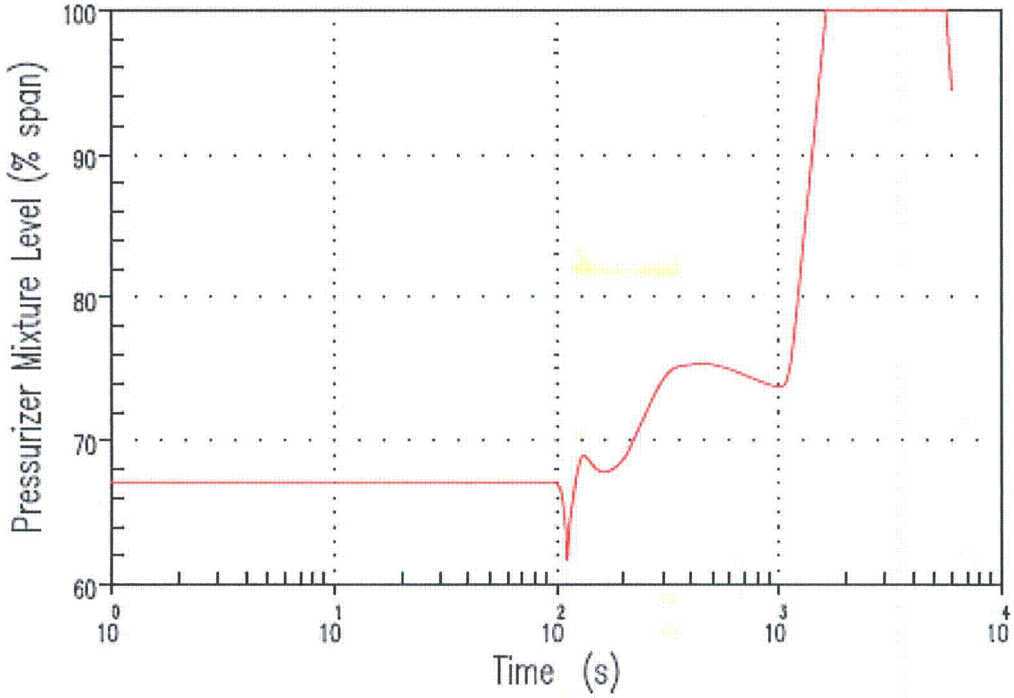


Figure SRXB-2.3.2-3 – Pressurizer Pressure vs. Time

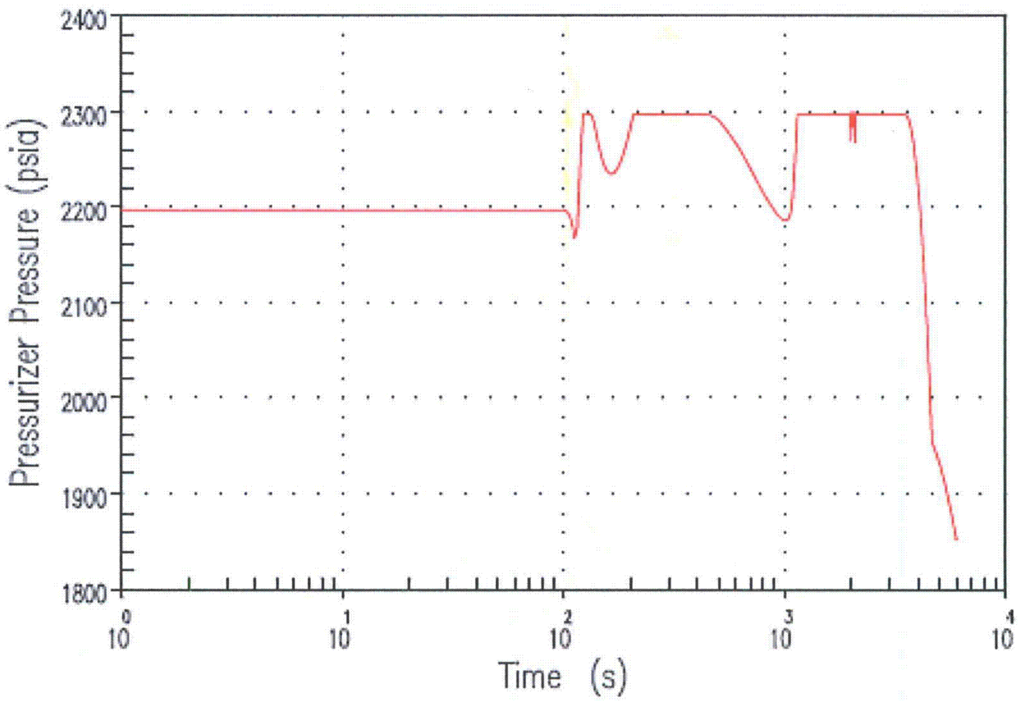


Figure SRXB-2.3.2-4 – Faulted Loop Temperatures vs. Time

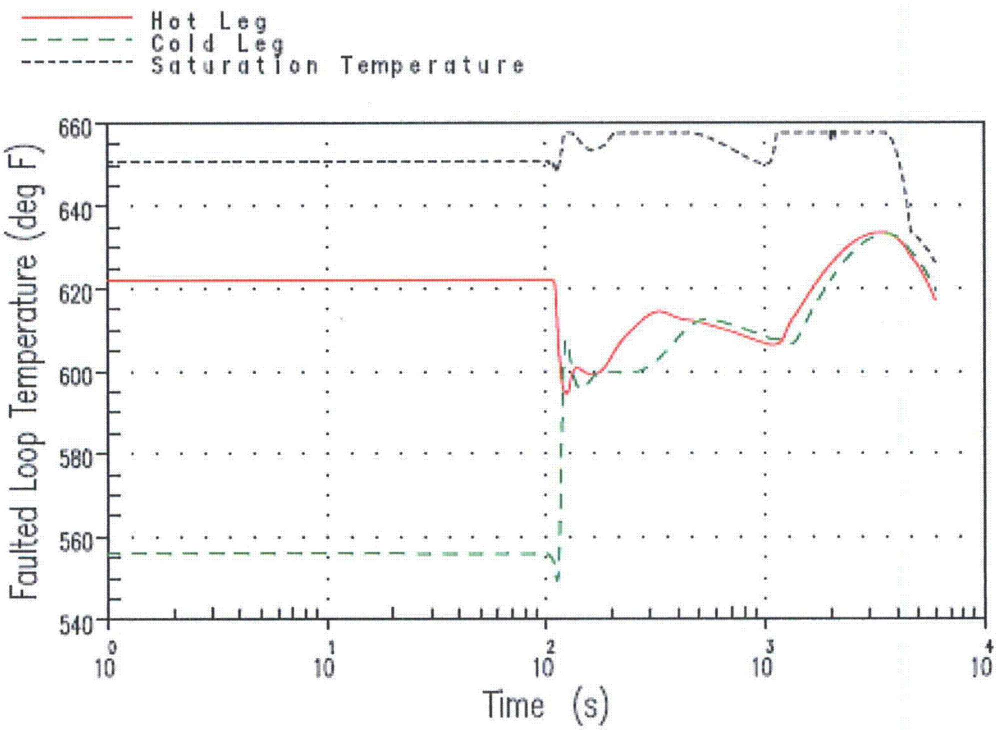


Figure SRXB-2.3.2-5 – Intact Loops Temperatures vs. Time

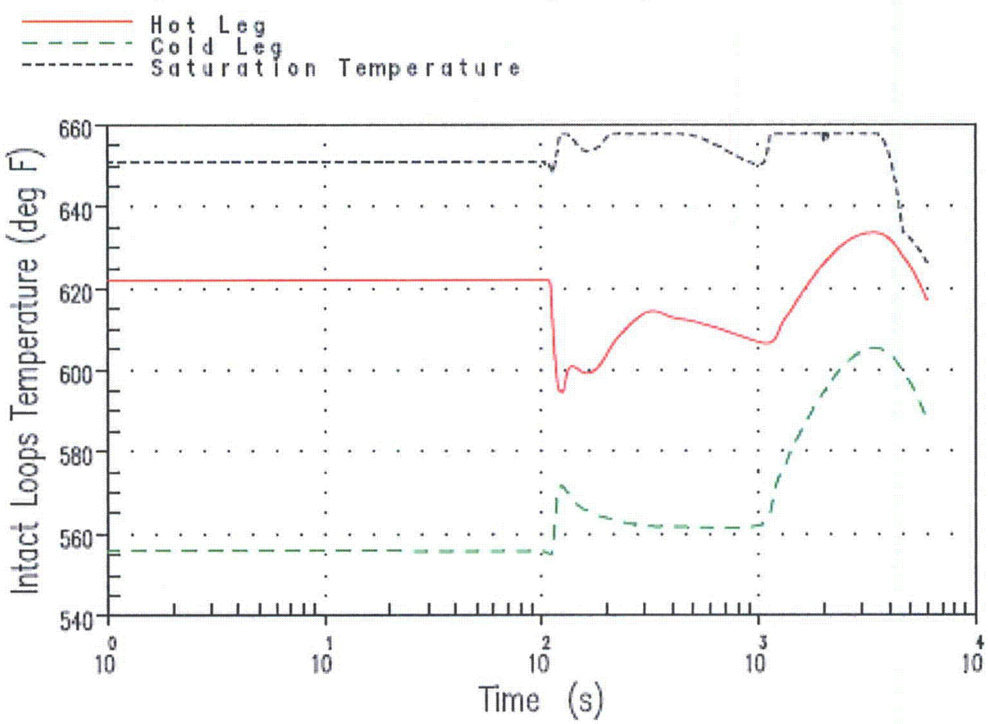


Figure SRXB-2.3.2-6 – Steam Generator Pressures vs. Time

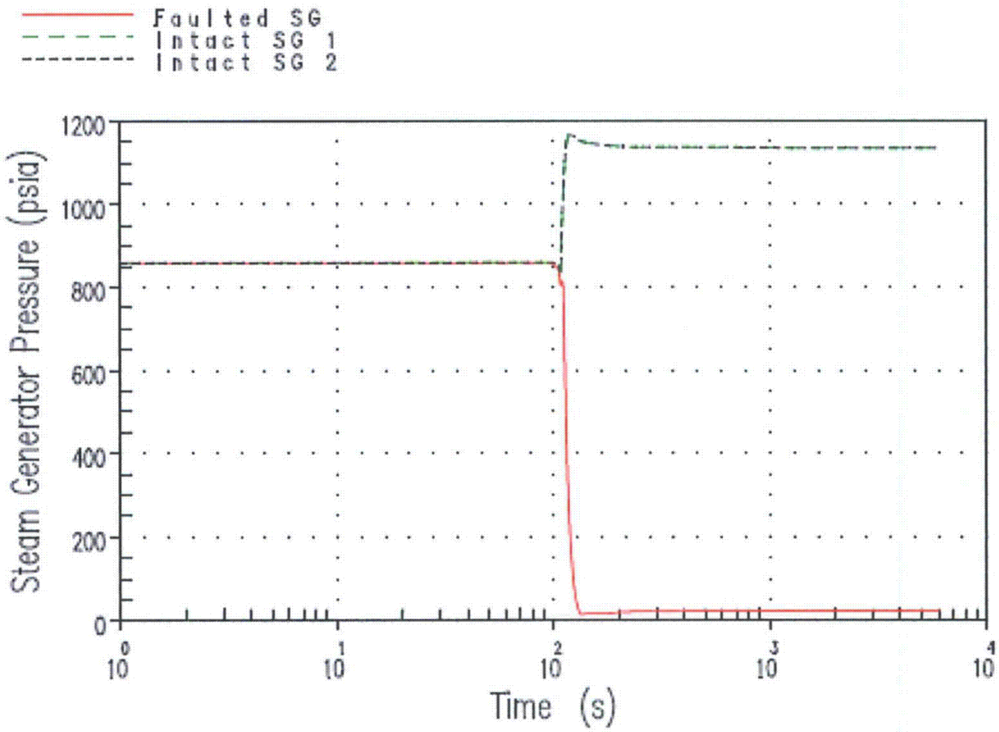


Figure SRXB-2.3.2-7 – Steam Generator Masses vs. Time

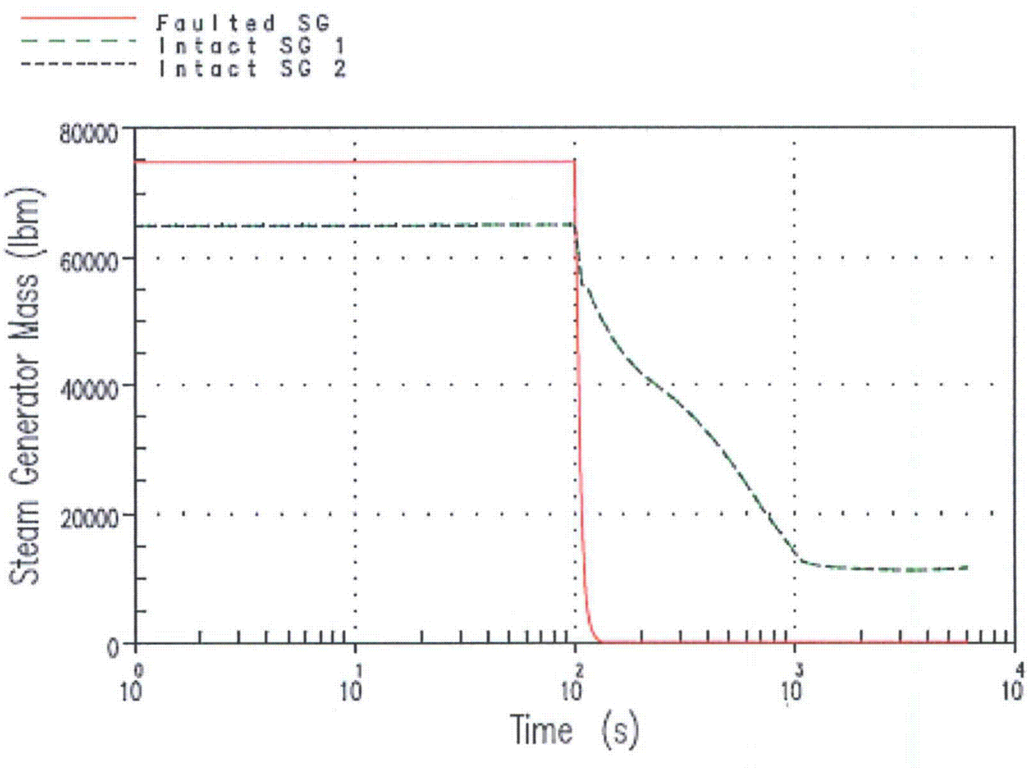


Figure SRXB-2.3.2-8 – Auxiliary Feedwater Flow vs. Time

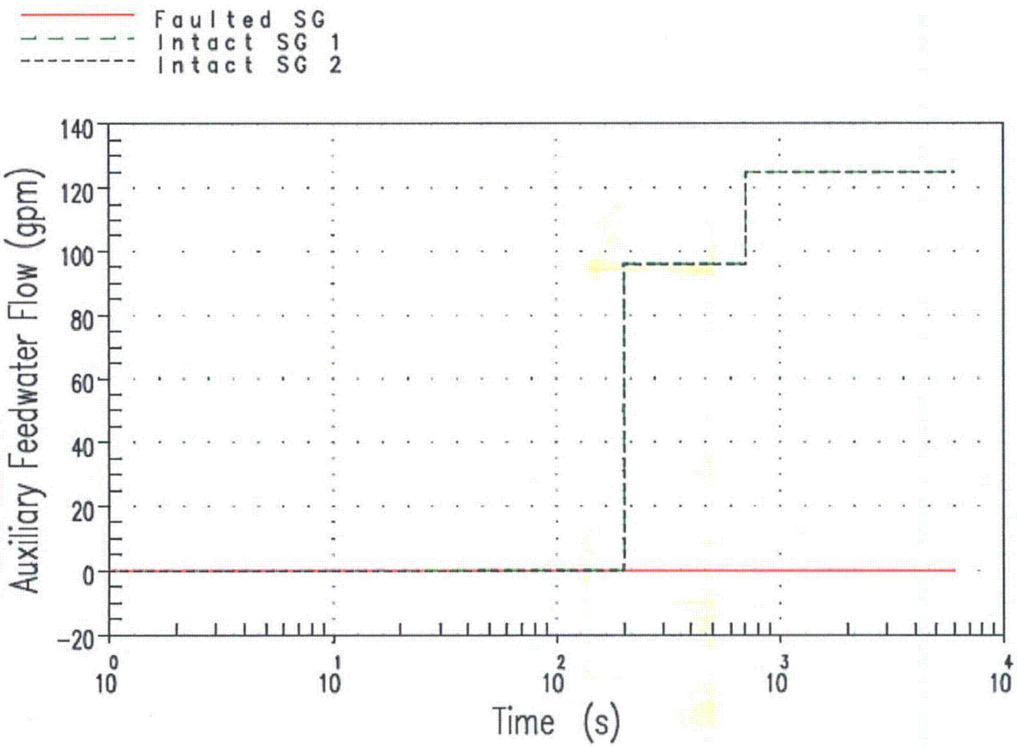
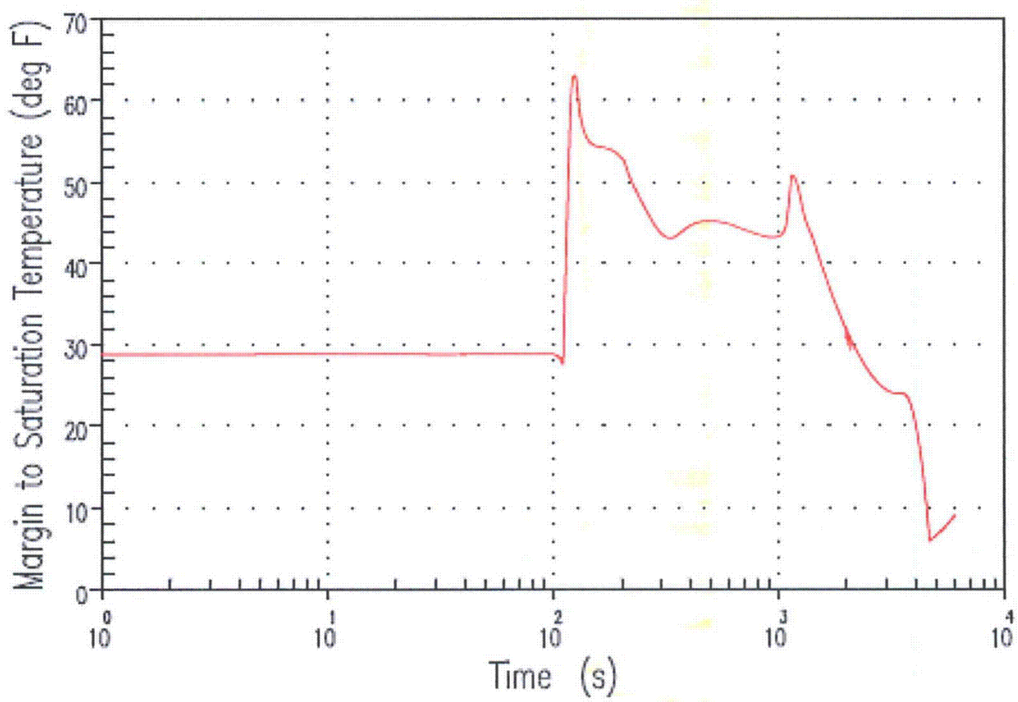


Figure SRXB-2.3.2-9 – Margin to Saturation Temperature vs. Time



References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.
2. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information – Round 1.3 (Part 3)", Accession No. ML11202A174, July 21, 2011.
3. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-233), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11221A227, August 5, 2011.
4. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information - Round 2.3 (Part 3)," Accession No. ML11263A204, September 20, 2011.
5. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-400), "Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205," Accession No. ML11276A080, September 30, 2011.

Turkey Point Units 3 and 4

RESPONSE TO NRC REACTOR SYSTEMS BRANCH REQUEST FOR ADDITIONAL INFORMATION REGARDING EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST NO. 205

ATTACHMENT 2

SRXB-1.3.21

Response to Request for Additional Information – Boron Dilution Analyses

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

By email dated July 21, 2011[Reference 2], the NRC Project Manager (PM) issued a Reactor Systems Branch (SRXB) RAI to FPL consisting of thirty-nine (39) questions on the loss of coolant accident (LOCA) and non-LOCA safety analyses presented in Reference 1. On August 5, 2011, FPL provided its response to RAIs SRXB-1.3.1-1.3.6 and SRXB-1.3.16-1.3.39 via FPL letter L-2011-233 [Reference 3].

RAI SRXB-1.3.21 of Reference 2 requested FPL to demonstrate that sufficient response time is available for operators to terminate a Boron Dilution event occurring in Modes 3, 4, or 5. The response to SRXB-1.3.21 maintained that a Boron Dilution event in Modes 3, 4, or 5 was beyond the current licensing basis for Turkey Point. During a follow-up telephone discussion between FPL and the NRC held on September 21, 2011, the SRXB staff requested that FPL supplement the response to SRXB-1.3.21 with specific analysis results generated for EPU conditions. The results of the requested analysis for Boron Dilution events in Modes 3, 4, and 5 are presented below.

In support of the new Boron Dilution analysis, Turkey Point is increasing the minimum required shutdown margin maintained in Mode 4 (without Reactor Coolant Pumps operating) and in Mode 5. Proposed amendments to the Turkey Point Technical Specifications (TS) reflecting these changes are included following the Boron Dilution analysis results.

SRXB-1.3.21 SRP Section 15.4.6 [1] lists the following acceptance criteria for B dilution event analyses:

If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:

A. During refueling: 30 minutes.

B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.

The applicant's analysis of the Chemical and Volume Control System Malfunction addresses only Modes 1, 2 and 6.

a. Provide analyses for this event in Modes 3, 4, and 5 (hot standby, hot shutdown, and cold shutdown, respectively). Initial conditions should consider the available shutdown margin, RCS pressure and charging flow, control rod positions and operability, available instrumentation and protective functions, and active RCS water volume (e.g., mid-loop operation) that are appropriate to each of these Modes.

Event Description:

Unborated water can be added to the reactor coolant system (RCS) via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction and cause an unwanted increase in reactivity and a decrease in shutdown margin. Plant operators should terminate an unplanned dilution before the shutdown margin is lost.

Based on Table 1.2 of the Turkey Point Units 3 and 4 TS, Operational Mode 3 (Hot Standby), Mode 4 (Hot Shutdown), and Mode 5 (Cold Shutdown) are all subcritical modes of operation that are distinguished from one another by the value of the average coolant temperature, T_{avg}, as follows.

- Mode 3: $T_{avg} \geq 350°F$
- Mode 4: $200°F < T_{avg} < 350°F$
- Mode 5: $T_{avg} \leq 200°F$

For each mode of operation, the TS define the requirements for the availability of the reactor coolant loops and/or residual heat removal (RHR) system loops, which are used to determine the appropriate active mixing volumes for the boron dilution cases being analyzed. Per TS 3.4.1.2, for **Mode 3**, all three of the reactor coolant loops shall be operable and in operation when the reactor trip breakers are closed and two reactor coolant loops shall be operable with at least one loop in operation when the reactor trip breakers are open. Having one reactor coolant loop in operation (one RCP running) is adequate to include all three loops in the active mixing volume. Per TS 3.4.1.3, for **Mode 4**, at least two loops among the three reactor coolant loops and two RHR loops shall be operable and at least one of these loops shall be in operation. Having one reactor coolant loop in operation (one RCP running) is adequate to include all three loops in the active mixing volume. However, with one RHR loop in operation, the active mixing volume includes the reactor vessel without the upper head, the volume of the RHR system, and part of one reactor coolant loop. Per TS 3.4.1.4.1, for Mode 5 with reactor coolant loops filled (**Mode 5 filled**), at least one RHR loop shall be operable and in operation, and either one additional RHR loop shall be operable or the secondary side water level of at least two SGs shall be greater than 10% narrow range span. The active mixing volume for Mode 5 filled is the same as that for Mode 4 on RHR. Per TS 3.4.1.4.2, for Mode 5 with reactor coolant loops not filled (**Mode 5 drained**), two RHR loops shall be operable and at least one RHR loop shall be in operation. In Mode 5 drained, the reactor coolant level may be as low as the mid-plane of the reactor vessel nozzles (mid-loop level). In this case, the active mixing volume for Mode 5 drained includes the filled portions of the vessel and one loop, plus the volume of the RHR system. Also examined was the Mode 5 configuration where the reactor vessel water level is three feet below the reactor vessel head flange (**Mode 5 partly drained**); this level is approximately 46 inches above the mid-loop level.

<u>Method of Analysis:</u>

For each Mode 3, 4, and 5 boron dilution scenario (case) examined, the method of analysis consists of a hand calculation that determines the time from the start of the dilution to when all shutdown margin is lost. The applied acceptance criterion is that there is at least 15 minutes from the start of the dilution to when all shutdown margin is lost. The inputs required for each calculation include the maximum dilution flow rate, the initial and critical boron concentrations (the difference of which corresponds to the applicable shutdown margin), the active mixing volume, the temperatures of the dilutant and of the RCS, and the pressures of the dilutant and of the RCS. Conservative values for each input parameter were applied so as to minimize the resultant times to lose shutdown margin. Table SRXB-1.3.21-1 identifies the cases that were examined and the corresponding inputs.

Results:

Table SRXB-1.3.21-2 summarizes the results for all cases analyzed.

For Mode 3 and Mode 4 with one RCP in operation, a comparison of the initial and critical boron concentrations from Table SRXB-1.3.21-1 below indicates that the shutdown margin would be lost sooner in Mode 4 versus Mode 3, as the difference between the initial boron concentration and the critical boron concentration is less for Mode 4. However, the results in Table SRXB-1.3.21-2 below show that shutdown margin is lost sooner for Mode 3, and this is because the time to lose shutdown margin is directly proportional to the RCS fluid density, which is significantly lower in Mode 3 versus Mode 4.

Conclusion:

The results of the Modes 3, 4, and 5 boron dilution analyses confirm that there is at least 15 minutes of time between the start of a boron dilution transient and when all shutdown margin is lost.

The indications available to operators in the Unit 3 and Unit 4 Control Rooms provide additional assurance that a dilution event can be terminated in a timely manner. Trips, alarms, and indications that could alert an operator to a dilution event in Modes 3, 4, and 5 are listed below. Those required to be operable in a given Mode by TS are so noted:

- Indicated increase in Source Range Neutron Flux count rate (Modes 3, 4, & 5)
- Source Range Hi Flux Alarm / Reactor Trip (Modes 3, 4, & 5 with reactor trip breakers closed)
- Primary Water Flowmeter
- Primary Water Totalizer (Provides audible count rate with a distinct tone heard through the Control Room)
- Primary Water Makeup Flow Deviation Alarm
- Volume Control Tank Level Indication and Level Alarms

Table SRXB-1.3.21-1 – Turkey Point Units 3 and 4 Boron Dilution in Modes 3, 4, and 5 – Input Summary

<table>
<tr><th>Case ID</th><th>Mode(1)</th><th>Minimum Active Mixing Volume(2) (ft^3)</th><th>Minimum Initial Boron Concentration (ppm)</th><th>Maximum Critical Boron Concentration (ppm)</th><th>Minimum Shutdown Margin (%Δk/k)</th><th>Maximum Dilution Flow Rate(3) (gpm)</th></tr>
<tr><td>3-RCP</td><td>3</td><td rowspan="2">6987.0</td><td>1800</td><td>1700</td><td rowspan="2">1.00</td><td rowspan="6">150</td></tr>
<tr><td>4-RCP</td><td rowspan="2">4</td><td>1690</td><td rowspan="5">1600</td></tr>
<tr><td>4-RHR</td><td rowspan="2">3579.7</td><td>1778</td><td rowspan="4">1.77</td></tr>
<tr><td>5-Filled</td><td rowspan="3">5</td><td>1770</td></tr>
<tr><td>5-Drained</td><td>2951.0</td><td rowspan="2">1779</td></tr>
<tr><td>5-Partly Drained</td><td>3146.5</td></tr>
</table>

(1) For Modes 3, 4, and 5, respectively, the applied average coolant temperatures are 547°F (Pressure = 2250 psia), 350°F, and 200°F. The RCS pressures applied for Modes 3, 4, and 5 are 2250 psia, 2250 psia, and 14.7 psia, respectively.

(2) Descriptions of the active mixing volumes are as follows.

6987.0 ft^3 - Volume of the reactor vessel minus the upper head volume plus the volume of all three reactor coolant loops, accounting for 10% steam generator tube plugging (SGTP); reactor coolant mixing is assumed to be provided by one RCP.

3579.7 ft^3 - Volume of the reactor vessel minus the upper head volume plus the volume of the RHR system plus the volume of one cold leg from the CVCS charging connection to the reactor vessel plus the volume of one hot leg from the RHR connection to the reactor vessel; reactor coolant mixing is assumed to be provided by the RHR system.

2951.0 ft^3 - Volume of the reactor vessel drained to the mid-loop level plus the volume of the RHR system plus half the volume of one cold leg from the CVCS charging connection to the reactor vessel plus half the volume of one hot leg from the RHR connection to the reactor vessel; reactor coolant mixing is assumed to be provided by the RHR system.

3146.5 ft^3 - Volume of the reactor vessel drained to 3 feet below the reactor vessel head flange plus the volume of the RHR system plus the volume of one cold leg from the CVCS charging connection to the reactor vessel plus the volume of one hot leg from the RHR connection to the reactor vessel; reactor coolant mixing is assumed to be provided by the RHR system.

(3) The conditions of the dilution water were based on a temperature of 39°F and a pressure of 14.7 psia. The maximum dilution flow rate of 150 gpm bounds the makeup flow that is required to maintain pressurizer water level when letdown flow is maximized. Maximum letdown flow is 120 gpm, but there are additional reactor coolant losses that correspond to allowable reactor coolant system leakage (11 gpm) and reactor coolant pump seal leakoff (15 gpm) that have been considered. The maximum dilution flow of 252 gpm that was applied in the boron dilution analyses for operating Modes 1, 2, and 6 is a very conservative dilution flow that corresponds to three charging pumps in operation.

Table SRXB-1.3.21-2: Turkey Point Units 3 and 4 Boron Dilution in Modes 3, 4, and 5 – Results Summary

Case ID	Mode	Time to Lose Shutdown Margin (minutes)
3-RCP	3	15.01
4-RCP	4	17.15
4-RHR		16.94
5-Filled	5	17.35
5-Drained		15.02
5-Partly Drained		16.02

TECHNICAL SPECIFICATION CHANGES

Technical Specification 3/4.1.1 Boration Control – Shutdown Margin - T_{avg} > 200°F

Current TS

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to the applicable value shown in Figure 3.1-1.

Proposed TS

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to the applicable value shown in Figure 3.1-1.

Figure 3.1-1, "Required Shutdown Margin vs Reactor Coolant Boron Concentration," is revised as shown in Figures SRXB-1.3.21-1a and SRXB-1.3.21-1b to provide a separate curve representing the minimum allowable shutdown margin in Mode 4 when no reactor coolant pumps are operating.

Basis for the Change: An increase in the minimum allowable shutdown margin for Mode 4 without reactor coolant pumps operating is necessary to assure adequate operator response time is available to identify and terminate an inadvertent dilution event prior to loss of all shutdown margin.

Technical Specification 3/4.1.1 Boration Control – Shutdown Margin - $T_{avg} \leq$ 200°F

Current TS

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/k, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

Proposed TS

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to **1.77 % Δk/k.**

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than **1.77 % Δk/k,** immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to **1.77 % Δk/k**:

Basis for the Change: An increase in the minimum allowable shutdown margin for Mode 5 is necessary to assure adequate operator response time is available to identify and terminate an inadvertent dilution event in Mode 5 prior to loss of all shutdown margin.

See Figure SRXB-1.3.21-2 for indicated changes.

Technical Specification 3/4.1.2 Boration Control – Flowpaths - Operating

Current TS

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

a. With no boration source path from a boric acid storage tank OPERABLE,

   2. Restore the boration source path from a boric acid storage tank to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the boration source path from a boric acid storage tank to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With only one boration source path OPERABLE or the regenerative heat exchanger flow path to the RCS inoperable, restore the required flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two boration source paths to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

c. With the boration source path from a boric acid storage tank and the charging pump discharge path via the regenerative heat exchanger inoperable, within one hour initiate boration to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F and go to COLD SHUTDOWN as soon as possible within the limitations of the boration and pressurizer level control functions of the CVCS.

Proposed TS

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

a. With no boration source path from a boric acid storage tank OPERABLE,

   2. Restore the boration source path from a boric acid storage tank to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a **boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F** within the next 6 hours; restore the boration source path from a boric acid storage tank to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With only one boration source path OPERABLE or the regenerative heat exchanger flow path to the RCS inoperable, restore the required flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a **boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F** within the next 6 hours; restore at least two boration source paths to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

c. With the boration source path from a boric acid storage tank and the charging pump discharge path via the regenerative heat exchanger inoperable, within one hour initiate boration to a **boron concentration equivalent to the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F** and go to COLD SHUTDOWN as soon as possible within the limitations of the boration and pressurizer level control functions of the CVCS.

Basis for the Change: The minimum allowable shutdown margin in Mode 5 (Cold Shutdown) is increased to 1.77 % Δk/k as described above to assure adequate operator response time is available to identify and terminate an inadvertent dilution event prior to loss of all shutdown margin. The change to TS 3.1.2.2 removes the numeric value of minimum shutdown margin at 200°F to more clearly convey the objective of the Action Statement to borate to Cold Shutdown conditions in advance of reaching Mode 5 entry conditions.

See Figure SRXB-1.3.21-3 for indicated changes.

Technical Specification 3/4.1.2 Boration Control – Charging Pumps - Operating

Current TS

3.1.2.3 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within 6 hours; restore at least two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

Proposed TS

3.1.2.3 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a **boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F** within 6 hours; restore at least two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

Basis for the Change: The minimum allowable shutdown margin in Mode 5 (Cold Shutdown) is increased to 1.77 % Δk/k as described above to assure adequate operator response time is available to identify and terminate an inadvertent dilution event prior to loss of all shutdown margin. The change to TS 3.1.2.3 removes the numeric value of minimum shutdown margin at 200°F to more clearly convey the objective of the Action Statement to borate to Cold Shutdown conditions in advance of reaching Mode 5 entry conditions.

See Figure SRXB-1.3.21-4 for indicated changes.

Technical Specification 3/4.1.2 Boration Control – Borated Water Sources - Operating

Current TS

3.1.2.5 The following borated water sources shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours* and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

Proposed TS

3.1.2.5 The following borated water sources shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours* and borated to a **boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F**; restore the Boric Acid Storage System to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

Basis for the Change: The minimum allowable shutdown margin in Mode 5 (Cold Shutdown) is increased to 1.77 % Δk/k as described above to assure adequate operator response time is available to identify and terminate an inadvertent dilution event prior to loss of all shutdown margin. The change to TS 3.1.2.5 removes the numeric value of minimum shutdown margin at 200°F to more clearly convey the objective of the Action Statement to borate to Cold Shutdown conditions in advance of reaching Mode 5 entry conditions.

See Figure SRXB-1.3.21-5 for indicated changes. In addition to the markup described above, the TS page shown on Figure SRXB-1.3.21-5 also includes the TS markups that were submitted with the original EPU LAR [Reference 1].

NO SIGNIFICANT HAZARDS DETERMINATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazard if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

FPL proposes to revise Turkey Point Units 3 and 4 Technical Specification (TS) Limiting Conditions for Operation (LCO) 3.1.1.1, 3.1.1.2, 3.1.2.2, 3.1.2.3, and 3.1.2.5 and Surveillance Requirement (SR) 4.1.1.2. The revisions will increase the minimum required shutdown margin in Mode 4 (without reactor coolant pumps) and Mode 5 to ensure adequate operator response time is available to identify and terminate an inadvertent boron dilution event in Mode 4 or 5.

FPL has reviewed this proposed license amendment for Turkey Point Units 3 and 4 and determined that its adoption would not involve a significant hazards consideration. The bases for this determination are:

The proposed amendment does not involve a significant hazards consideration for the following reasons:

1. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The proposed changes to TS 3.1.1.1, TS 3.1.1.2, TS 3.1.2.2, TS 3.1.2.3, TS 3.1.2.5, and SR 4.1.1.2 increase the minimum shutdown margin required to be maintained in Mode 4 (without reactor coolant pumps operating) and in Mode 5. While an inadvertent boron dilution event occurring in Mode 4 or Mode 5 has not been previously analyzed, these changes would increase the available time for operators to identify and terminate a dilution event before a complete loss of shutdown margin could occur. With the updated minimum shutdown margin limits, the analysis results discussed above assure that available operator action time is within the minimum acceptance criterion.

   The proposed changes have no effect on the probability of an accident previously evaluated as they do not affect the configuration or operation of systems that could initiate a dilution event. The proposed changes have no significant effect on the consequences of a dilution event as they only provide increased assurance that a loss of shutdown margin can be prevented should an inadvertent dilution event occur. Radiological consequences of a boron dilution event are not evaluated since shutdown margin is shown to be maintained.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The proposed TS and SR changes increase the minimum shutdown margin required to be maintained in Mode 4 (without reactor coolant pumps operating) and in Mode 5. The TS and SR changes do not impact the operation of the reactor coolant system (RCS) or the supporting systems involved in initiating or terminating a dilution event. The operation of plant equipment in any Mode is unaffected by this change.

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

3. **The proposed amendment does not involve a significant reduction in the margin of safety.**

   The proposed TS and SR changes provide the operators with increased available response time to terminate an inadvertent dilution event, ensuring that shutdown margin can be maintained in Mode 4 (without reactor coolant pumps operating) and in Mode 5. As this change affords more time for the operators to respond and terminate the dilution event, an increase in the margin of safety occurs.

   Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above discussion, FPL has determined that the proposed change does not involve a significant hazards consideration.

Figure SRXB-1.3.21-1a

DELETE THIS FIGURE. REPLACE WITH FIGURE SRXB-1.3.21-1b ON NEXT PAGE.

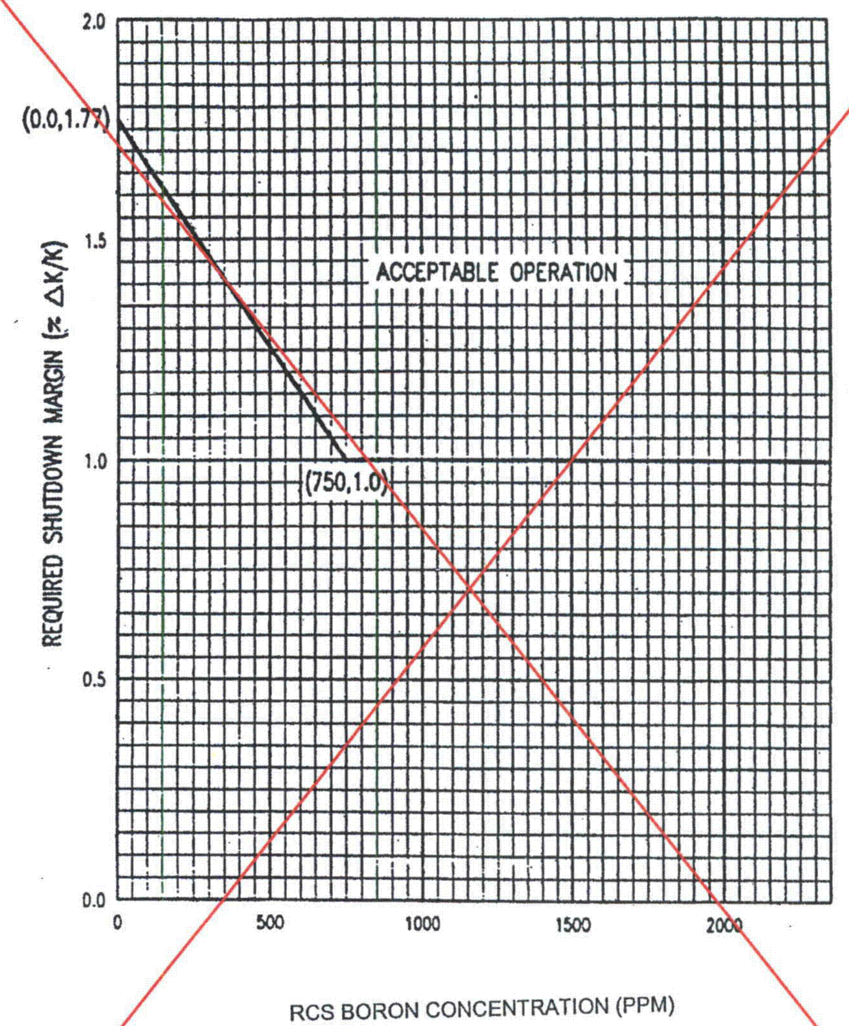


Figure 3.1-1
Required Shutdown Margin vs Reactor Coolant Boron Concentration

Figure SRXB-1.3.21-1b

INSERT FOR FIGURE 3.1-1

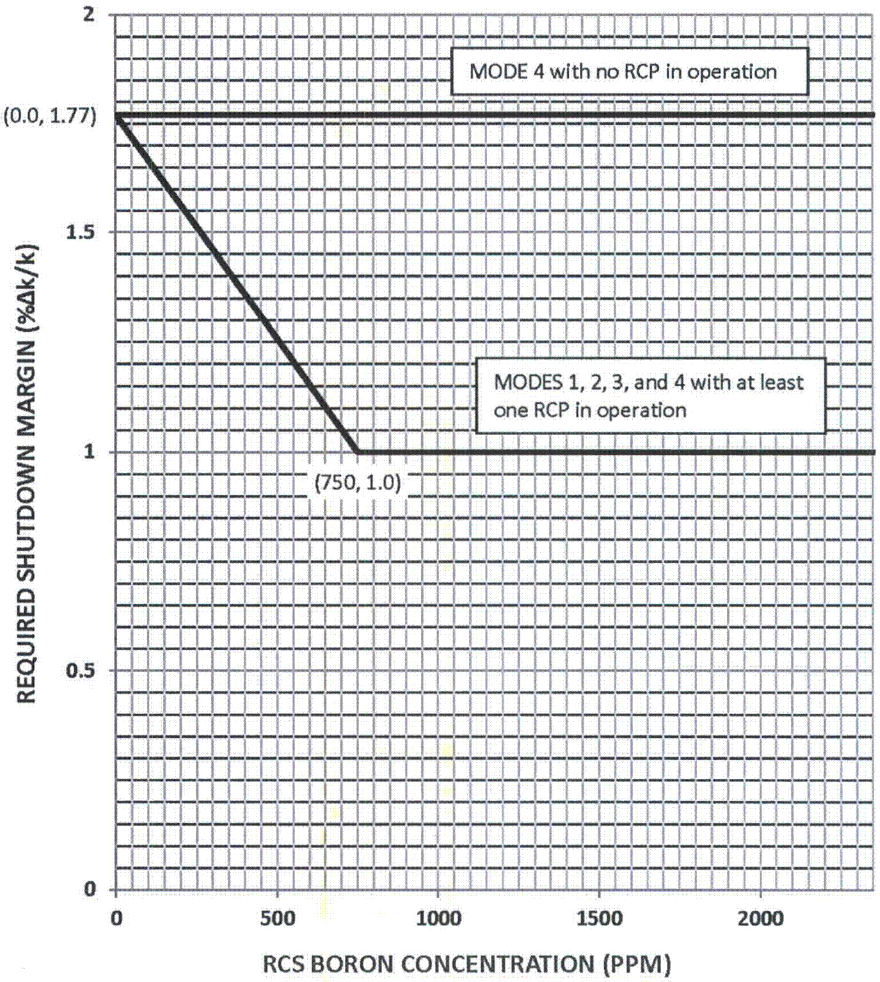


Figure 3.1-1
Required Shutdown Margin vs Reactor Coolant Boron Concentration

Figure SRXB-1.3.21-2

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

1.77%

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to ~~1%~~ Δk/k.

APPLICABILITY: MODE 5.

1.77%

ACTION:

With the SHUTDOWN MARGIN less than ~~1%~~ Δk/k, immediately initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

1.77%

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1%~~ Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

Figure SRXB-1.3.21-3

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

a. The source path from a boric acid storage tank via a boric acid transfer pump to the charging pump suction*, and

b. At least one of the two source paths from the refueling water storage tank to the charging pump suction; and,

c. The flow path from the charging pump discharge to the Reactor Coolant System via the regenerative heat exchanger.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200 °F

a. With no boration source path from a boric acid storage tank OPERABLE,

1. Demonstrate the OPERABILITY of the second source path from the refueling water storage tank to the charging pump suction by verifying the flow path valve alignment; and

2. Restore the boration source path from a boric acid storage tank to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a ~~SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F~~ within the next 6 hours; restore the boration source path from a boric acid storage tank to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With only one boration source path OPERABLE or the regenerative heat exchanger flow path to the RCS inoperable, restore the required flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a ~~SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F~~ within the next 6 hours; restore at least two boration source paths to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

c. With the boration source path from a boric acid storage tank and the charging pump discharge path via the regenerative heat exchanger inoperable, within one hour initiate boration to a ~~SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F~~ and go to COLD SHUTDOWN as soon as possible within the limitations of the boration and pressurizer level control functions of the CVCS.

boron concentration equivalent to the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200 °F

* The flow required in Specification 3.1.2.2.a above shall be isolated from the other unit from the boric acid transfer pump discharge to the charging pump suction.

Figure SRXB-1.3.21-4

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200 °F

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a ~~SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F~~ within 6 hours; restore at least two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The required charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

Figure SRXB-1.3.21-5

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.5 The following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume in accordance with Figure 3.1-2,

2) A boron concentration in accordance with Figure 3.1-2. and

3) A minimum boric acid tanks room temperature of ~~55°F~~.

b. The refueling water storage tank (RWST) with: 62°F

1) A minimum indicated borated water volume of 320,000 gallons,

2) A ~~minimum~~ boron concentration ~~of 1950 ppm~~. between 2400 ppm and 2600 ppm

3) A minimum solution temperature of 39°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours* and borated to a ~~SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F~~; restore the Boric Acid Storage System to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the boric acid tank inventory concentration greater than ~~3.5~~ wt%, verify that the boric acid solution temperature for boration sources and flow paths is greater than the solubility limit for the concentration.

4.0

boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200 °F

* If this action applies to both units simultaneously, be in at least HOT STANDBY within the next twelve hours.

References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.
2. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information – Round 1.3 (Part 3)", Accession No. ML11202A174, July 21, 2011.
3. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-233), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Systems Issues," Accession No. ML11221A227, August 5, 2011.