FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT C

AREVA DOCUMENT ANP-3156(NP) – CRYSTAL RIVER 3 EPU BORIC ACID PRECIPITATION RAI RESPONSES (NON-PROPRIETARY)

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Crystal River 3 EPU
Boric Acid Precipitation RAI Responses

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List of Commonly Used Acronyms

BAP	Boric Acid Precipitation
BAST	Boric Acid Storage Tank
B&W	Babcock and Wilcox
BWST	Borated Water Storage Tank
CL	Cold Leg
CLPD	Cold Leg Pump Discharge
CLPS	Cold Leg Pump Suction
CR-3	Crystal River Unit 3
DC	Downcomer
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
FPC	Florida Power Corporation
HLI	Hot Leg Injection
HL-BD	Hot Leg Blowdown
HPI	High Pressure Injection
ITS	Improved Technical Specification
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LOCA	Loss-of-Coolant Accident
LP	Lower Plenum
LPI	Low Pressure Injection
LTCC	Long-Term Core Cooling
MWt	Megawatts
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
RAI	Request for Additional Information
RBS	Reactor Building Spray
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Vessel
RVC	Reactor Vessel Nozzle Centerline
RVVV	Reactor Vessel Vent Valve
SBLOCA	Small Break Loss of Coolant Accident
SG	Steam Generator
SNPB	Nuclear Performance and Code Review Branch
UP	Upper Plenum

1.0 Introduction

By letter dated June 15, 2011, Florida Power Corporation (FPC) requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt (Reference 1). On May 8, 2012, via electronic mail, the Nuclear Regulatory Commission (NRC) provided a draft request for additional information (RAI) needed to support the Nuclear Performance and Code Review Branch (SNPB) technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR). During the June safety analysis audit and by teleconference on June 21, 2012, CR-3 discussed the draft RAI with the NRC to confirm an understanding of the information being requested. On August 2, 2012, the NRC provided formal RAIs (Reference 2) required to complete its evaluation of the CR-3 EPU LAR (ML12202A060).

The NRC RAIs and the responses are provided in this document. This material contains Proprietary Information to be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390(a)(4). Proprietary Information is identified with brackets "[ ]".

For tracking purposes, each item related to this RAI is uniquely identified as SNPB X-Yz, with X indicating the RAI set (in this case 2) and Yz indicating the sequential item number.

2.0 Background

The CR-3 EPU LAR described various plant modifications implemented to support the power uprate. One of the modifications addresses post-LOCA boric acid precipitation (BAP) at the EPU power level. This modification includes a connection to the new low pressure injection (LPI) cross-tie pipe that attaches to the Reactor Coolant System (RCS) side of the decay heat drop line isolation valves. This single line branches into two pipes at one location, and each branch contains an isolation valve energized by different trains of power. This design is single failure proof to open. When at least one of these valves is opened by the operators from the main control room, sufficient hot leg injection (HLI) flow is provided to dilute the core boric acid concentration for a cold leg pump discharge (CLPD) break when the RCS pressure is less than or equal to 140 psia. Analyses have been performed to determine the time for operator action to open these valves as a function of RCS pressure when the core exit is inadequately subcooled. CR-3 also elected to add an alternative boric acid dilution option to manage core boron concentrations for accidents that remain above 140 psia in the RCS. This alternate boric acid dilution method is referred to as the hot leg blowdown (HL-BD) option. It works by initiating a forward flow through the core and out the decay heat drop line to the reactor building floor via one of two pipe branches connected to the RCS side of the HLI line. This blowdown line has two branches each with dual isolation valves that ensure that one branch line can open regardless of the single failure considered. When opened, one line can provide sufficient flow to dilute the core boron concentration at any RCS pressure. This second plant modification provides additional diversity to the boric acid dilution strategies and can be initiated to provide core boric acid dilution at higher RCS pressures.

3.0 NRC RAIs with Responses

This section repeats the formally transmitted NRC RAIs and provides a response to the questions.

SNPB 2-1: Please provide the following information for the CR-3 Nuclear Steam Supply System (NSSS):

SNPB 2-1a: Volume of the lower plenum, core and upper plenum below the bottom elevation of the hot leg, each identified separately. Volume in the downcomer below the bottom of the discharge leg. Also provide heights of these regions.

Response:

One of the key inputs to the CR-3 BAP analyses is the fluid volumes and the heights of those volumes in the downcomer, lower plenum, core, and upper plenum regions.

Following core quench, the excess emergency core cooling system (ECCS) flow refills the reactor vessel with water during the long-term core cooling (LTCC) phase of the cold leg loss-of-cooling accident (LOCA). The water level in the downcomer rises to the bottom of the cold leg pipe because ECCS injection exceeds the flow needed to match core boil-off. The excess ECCS refills the downcomer and a portion of the cold leg discharge pipe inlet nozzle and spills the excess liquid out of the break. The CR-3 BAP analyses used a water level in the downcomer region refilled to eight inches above the bottom of the cold leg pipe. The location of the bottom of the downcomer does not have a clear geometrical boundary at which to stop the fluid volume. For this response, the bottom of the downcomer is defined as an extension of the annular flow area down to the bottom of the reactor vessel lower plenum flow distributor plate. The total fluid volume from six inches below the cold leg nozzle centerline to the bottom of the flow distributor plate is [ ]. These values are shown on the simplified sketch in Figure 1.

The RAI specifically requests the water volume up to the bottom of the cold leg. The cold leg pipe inside diameter is 28 inches. Based on a downcomer area of [ ].

The lower plenum is the volume from the bottom of the reactor vessel up to bottom of the unheated core region, excluding the downcomer annulus extension. It has a volume of [

].

The CR-3 BAP analyses combine the downcomer with the lower plenum fluid volume into a combined "downcomer" region. It has a combined liquid volume when filled to six inches below the cold leg nozzle centerline of [

]

The core region is defined to include both the active core region as well as the unheated regions above and below the core. The total core region is [

], respectively. The baffle region is the area between the core baffle plates and the cylindrical core barrel, which divides the downcomer from the core region. The core bypass is the fluid volume inside the guide tubes minus the volume occupied by inserted control rods.

The key elevations and volumes for BAP in the 177 fuel assembly Babcock and Wilcox (B&W)-designed plants are based on the plenum cylinder height of the upper plenum and outlet annulus below the bottom of the reactor vessel vent valves (RVVVs). The bottom of the RVVVs is at approximately the same elevation as the bottom of large holes in the plenum cylinder. These large holes limit the two phase mixture swell to a level slightly above the bottom of the large holes in the plenum cylinder. This region is [

] for the upper plenum and outlet annulus regions, respectively.

The RAI asked for the volume of the upper plenum fluid below the bottom of the hot leg pipe. The length from the top of the core region to the bottom of the hot leg is [

] It should be noted that volume of the upper plenum fluid below the bottom of the hot leg pipe is not a key characteristic of B&W plants because of the plenum cylinder and the RVVVs, so it is not specifically modeled.

Figure 1: Reactor Vessel Region Volumes

SNPB 2-1b: Loop friction and geometry pressure losses from the core exit through the steam generators to the inlet nozzle of the reactor vessel. Also, provide the locked rotor reactor coolant pump (RCP) k-factor. Please provide the mass flow rates, flow areas, k-factors, and coolant temperatures for the pressure losses provided (upper plenum, hot legs, steam generators (SGs), suction legs, RCPs, and discharge legs). Please include the reduced SG flow areas due to plugged tubes. Please also provide the loss from each of the intact cold legs through the annulus to a single broken cold leg. Please also provide the equivalent loop resistance in feet/gallons per minute^2 (ft/gpm^2).

Response:

The loop and RCP resistance information is not a key characteristic for the BAP analysis of a B&W-designed plant because of the RVVV operation. The CR-3 BAP analyses conservatively assume the cold leg pump suction (CLPS) piping contains liquid and as such no steam flow through the loops is credited. While the loop seals will be cleared during the blowdown phase for the largest LOCAs, they will not clear for smaller LOCAs. It is also possible that loop seals could reform during the LTCC phase, so the entire suite of CR-3 BAP analyses did not credit any flow through the loops. Therefore, this information is not provided because the equivalent loop resistance is modeled as effectively infinite.

SNPB 2-1c: Number of vent valves, flow areas and geometric k-factor. Elevation and diameter of the vent valves.

Response:

There are eight 14" inside diameter RVVVs with a total combined area of 8.55 ft^2 and a bottom elevation roughly 3 ft above the reactor vessel nozzle centerline. These inertial flapper valves begin to open or lift off their seats at [

]

[

]

Where

ΔP = pressure difference from the inside to the outside of the RVVVs in lbf/in^2

(Note that the ΔP must be converted to lbf/ft^2 to resolve the units.)

K_{RVVV} = RVVV form loss

A_{RVVV} = RVVV flow area in ft^2

W_{RVVV} = RVVV mass flow rate in lbm/s

ρ_{RVVV} = RVVV inlet fluid density in lbm/ft^3

g_c = 32.174 lbf-ft/lbm-s^2

SNPB 2-1d: Friction and geometric total resistance from the top of the core through the vent valves to the discharge leg. Please also provide the equivalent resistance for this path in ft/gpm^2.

Response:

The RAI asks for the resistance in the units of ft/gpm^2. The information provided gives the resistance in the form of k/A^2, which when used with the Bernoulli formulation, calculates the differential pressure versus flow and density. This equation does not require translation of the resistances when the fluid is two-phase or single phase steam.

The resistance requested was separated into three components: the RVVVs, the entrance into the broken cold leg nozzle, and the resistance from the top of the core to the RVVVs through the large holes in the plenum cylinder.

The RVVVs provide the largest form loss resistance for this combined path [ ] using the area and the form loss equation listed in the response to RAI SNPB 2-1c. The next largest form resistance to flow from the top of the core to the break is the cold leg nozzle reverse loss. The turning loss from the upper downcomer into the cold leg and its area contraction has a [ ] The resistance from the core exit, through the large plenum cylinder holes to the RVVVs is small in comparison to these other two resistances. This combined loss is nearly two orders of magnitude less than the RVVV and cold leg nozzle contributions, so core exit and large hole resistance is neglected along with the frictional losses for any part of this flow path.

So the net resistance from the top of the core to the broken discharge leg is effectively approximated by using the RVVVs and the cold leg nozzle entrance losses. These two resistances can be added to obtain the overall form loss net resistance from the top of the core into the broken cold leg.

Note that the RVVV resistance provided is for a full open RVVV. The RVVV differential pressure will be less than the full open value so the RVVV will not be full open and the resistance will be higher as calculated based on the RVVV form losses provided in SNPB 2-1c.

SNPB 2-1e: Capacity and boron concentration of the refueling water storage tank and core flood tanks.

Response:

CR-3 refers to the refueling water storage tank as the borated water storage tank (BWST). The CR-3 BWST can contribute a maximum volume of 383,198 gallons to these analyses. This is the volume between the maximum Improved Technical Specification (ITS) level (which equates to 449,000 gallons) to the low-low alarm level (7 feet which corresponds to completion of sump swap-over) and protects a vortex formation level of 5.5 feet. The maximum ITS boron concentration is 3000 ppm.

There are two core flood tanks with nominally contain 1080 ft^3 per tank including the line liquid volume. The ITS maximum boron concentration is 3500 ppm but is modeled with a conservative value of 4000 ppm.

SNPB 2-1f: Capacity of the condensate storage tank

Response:

The CR-3 BAP analysis does not rely on RCS cool-down so this is not a necessary input to the analyses performed by AREVA. Nevertheless, the maximum usable volume in the condensate storage tank is 178,238 gallons. The primary source of emergency feedwater at CR-3 is a dedicated Emergency Feedwater Tank with a capacity of 150,000 gallons backed up by the condensate storage tank and two Fire storage tanks each with ~ 330,000 gallons of usable volume.

SNPB 2-1g: Boric acid concentration versus time for the limiting large break.

Response:

A range of break sizes is considered for the CR-3 BAP analyses. The double-ended guillotine break in the cold leg pump discharge piping has the largest void fractions in the core and upper plenum region and, therefore, has the smallest core mixing volume. Without credit for RVVV liquid overflow or hot leg nozzle gap flows, the double-ended guillotine break produces the fastest boron concentration increase and requires the earliest operator action times to initiate active boron dilution measures. Its concentration versus time is provided in response to SNPB 2-5 in Figure 4. The RCS pressure is conservatively assumed to be 14.7 psia and credit for operator initiation of 400 gpm of HLI is included in this analysis at 2.5 hours.

SNPB 2-1h: Flushing flow rate at the time of switch to simultaneous injection.

Response:

The CR-3 plant has developed two methods to provide active boron dilution. Each is single failure-proof and the methods are independent of one another. One credits opening of at least one of the isolation valves to provide HLI from a fraction of the net LPI flow. The portion of the LPI flow not provided to the HLI path enters the reactor vessel downcomer through the core flood nozzles. The other is a boron dilution HL-BD line that initiates forward flow through the core.

The HLI option provides flows into the decay heat removal line and then into the bottom of the hot leg horizontal piping before being delivered to the top of the core. This is in the opposite direction of normal decay heat removal flow. The HLI option provides more than 400 gpm with one LPI pump in operation when the RCS pressure is less than or equal to 140 psia. The HLI flow increases at RCS pressures below 140 psia or when two LPI pumps are in operation. For conservatism, only 400 gpm of HLI flow was credited in the CR-3 BAP analyses when the RCS pressure was 140 psia or less. When the RCS pressure is higher than 140 psia, the HLI flow rapidly decreases and it is conservatively assumed to be zero. For break sizes that remain at pressures above 140 psia, the second boron dilution HL-BD option is used.

The HL-BD option allows flow from within the hot leg horizontal piping to flow into the bottom of the containment building through a pipe that connects to the HLI line. The HLI line connects into the decay heat drop line on the RCS side of the decay heat drop line isolation valves.

The HLI initiates a reverse core flow while the HL-BD initiates a forward core flow. The HL-BD flow rate is a function of RCS pressure, but it will provide more than 50 gpm when the RCS and reactor building are at atmospheric pressure with only the elevation head driving the flow. Either the HLI or HL-BD boron dilution flow paths effectively dilute the core concentration when initiated before the solubility limit is reached and the RCS pressure is less than or equal to 140 psia. The HL-BD flows will dilute the core concentration when the RCS pressure is above the HLI pressure range. Hydraulic analysis have been performed to demonstrate that the LPI pumps deliver the required LPI and HLI flows simultaneously with the as-designed LPI cross-tie and HLI flow paths.

SNPB 2-1i: High pressure safety injection runout flow rate

Response:

Each high pressure injection pump contains a stop check valve with the design function and capability to be throttled to prevent runout or exceeding a flow rate of 600 gpm used in the diesel loading calculations. Out of the 3 high pressure injection (HPI) pumps, only one is currently required to be throttled.

SNPB 2-1j: Please provide the limiting top skewed and bottom skewed axial power shapes (axial peaking factor versus axial position). Please provide a numerical table of this information.

Response:

The core inlet and core exit 1.7 axial power shapes are provided in Tables 1 and 2, respectively. The core inlet shape is used to define the limiting core mixing volumes. The core exit peak is used for the small break Loss-of-Coolant Accident (SBLOCA) PCT analyses, but it is not used in the CR-3 BAP analyses.

Table 1: Core Inlet (2.506-ft) Axial Power Shape

[ ]

[ ]

Table 2: Core Exit (10.811-ft) Axial Power Shape

[ ]

[ ]

SNPB 2-1k: Bottom elevation of the suction leg horizontal leg piping.

Response:

The elevation of the bottom of the CLPS horizontal piping is **[ ]** below the reactor vessel nozzle centerline. This is shown in Figure 2. This elevation may not be essential since flow through the loop seals is not credited based on the steam flow through the RVVVs.

Figure 2: Layout of Key Reactor Vessel Internals

SNPB 2-1l: Top elevation of the cold leg at the RCP discharge.

Response:

The centerline elevation of the RCP discharge piping is [ ] above the reactor vessel nozzle belt centerline. The pipe is 28 inches in diameter, so the top of the pipe is at [ ] above the reactor vessel nozzle belt centerline. This is shown in Figure 2.

SNPB 2-1m: Top elevation of the core (also height of core).

Response:

The top of the heated core is [ ] below the reactor vessel nozzle belt centerline. The heated core height is modeled to be 12 ft high. The core region (including the core bypass and baffle regions as well as the unheated upper and lower core heights) modeled in the CR-3 BAP analyses is [ ] below the reactor vessel nozzle belt centerline, respectively. These unheated core regions are the upper and lower portions of the fuel pins without uranium plus part of the end fittings. Figure 1 and Figure 2 show these heights or provide sufficient information to calculate them.

SNPB 2-1n: Bottom elevation of the downcomer.

Response:

In the RCS system models, the bottom of the downcomer annulus is at the bottom of the lower plenum flow distributor plate. This elevation is [ ] below the reactor vessel nozzle belt centerline as shown in Figure 2.

SNPB 2-2: What is the sump temperature versus time following recirculation initiation and how does this impact precipitation? Is the boric acid concentration in the vessel below the precipitation limit based on the minimum sump temperature at the time the switch to simultaneous injection is performed? Please explain.

Response:

During sump recirculation, the fluid from sump is cooled by the decay heat cooler prior to its injection into the RCS. The ECCS injection temperature is the key temperature that is used in the CR-3 BAP analyses. This time-dependent fluid temperature is a function of break size, break location, time post-trip, number of ECCS and reactor building spray (RBS) pumps in operation, and ultimate heat sink temperature. The highest sump recirculation ECCS temperature is expected to be less than 185 F for the cold leg pump discharge breaks, but a higher value was used to provide conservatism in the analyses. A conservatively high constant temperature of 200 F is used for the analyses to determine the required HLI flow rate to match core decay heat. Although the ECCS injection temperature decreases with time, the ECCS inlet temperature was held at a high constant as a simplification when injection is coming from the sump. This high inlet temperature also decreases the core inlet subcooling and reduces the core mixing volume.

A lower ECCS injection temperature could potentially increase the likelihood of the mixed fluid from the core and downcomer exceeding the solubility limit in the lower plenum region. However, the ECCS injection entering the upper downcomer and/or HPI lines condenses steam flowing through the RVVVs and mixes with the fluid in the downcomer liquid region before it reaches the lower plenum of the reactor vessel. As the downcomer fluid enters the lower plenum its temperature has been increased above the ECCS injection temperature so the temperature and solubility limit of this fluid is higher. When this fluid mixes with fluid from the core that is hotter and at higher boron concentrations, the solubility limit of this mixed fluid is evaluated to ensure precipitation does not occur. A similar check is performed in the mixed fluids in hot leg once HLI is initiated as well. The solubility limit in either the lower plenum or the hot leg will not exceed the solubility limit when a mixing limit is imposed. By imposing a mixing limit on the calculations, the solubility limit in the lower plenum will not be violated.

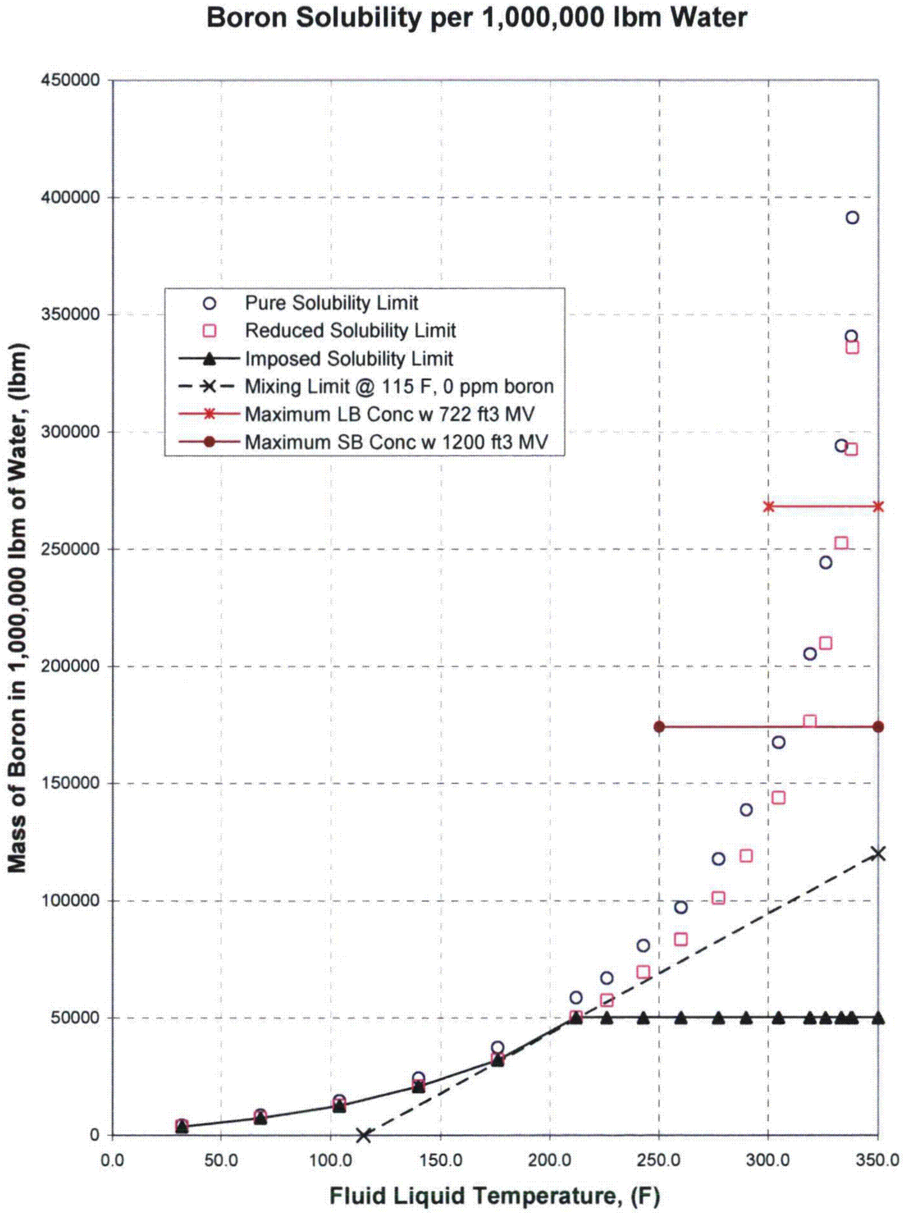
The CR-3 BAP analysis evaluates a mixing limit between fluids combined at different temperatures and boron concentrations. When these fluids are mixed, the resultant temperature and concentration is the average temperature and concentration of the two fluid components based on their respective mass contributions. The resultant temperature and concentration fall on the line connecting the colder dilute fluid and the higher temperature concentrated fluid. Should this mixed condition exceed the solubility limit there could be precipitation. Precipitation from mixing with a cold fluid is precluded by including a mixing limit in the BAP analyses.

The mixing limit is line that begins with the concentration and fluid temperature from HLI (for the hot leg) or the downcomer (for the lower plenum) and this line is a tangent to the saturation temperature solubility limit in the core region. This mixing limit is compared to the mixed-mean temperature and concentration of the two fluid components. When the mixed-mean conditions are below the mixing limit, no precipitation can occur.

Figure 3 provides an example in which 115 F water with a zero boron concentration is injected into the hot leg following HLI initiation. The mixing limit for this fluid temperature is the line that goes through this temperature and is tangent to the reduced solubility limit, which is the pure boric acid solubility limit reduced by the fractional equivalent of 4 w% H_3BO_3. In this case, the mixing limit is more restrictive for temperatures less than 212 F. At higher temperatures, the constant solubility limit equivalent to that for a saturated mixture at 212 F, which is imposed in the CR-3 EPU BAP analyses, is more limiting than the mixing limit. The minimum of the mixing limit in the hot leg or lower plenum and the constant saturated solubility limit in the core determines when an active dilution method is needed. The saturated solubility limit prevents precipitation in the core. The mixing limits prevent precipitation in the hot leg or the lower plenum regions.

Also shown on Figure 3 are the core concentrations based on putting all available boron mass (BWST, RCS, CFT, and BAST injection) in the core region with a minimum LBLOCA core mixing volume of 722 ft^3. A similar concentration calculation was performed for the SBLOCA core mixing volume of 1200 ft^3 and it is also shown on the figure. These two lines represent the maximum core concentration that the core could achieve after several weeks of core boiling without any active or passive boron dilution methods providing boron dilution.

Figure 3: Boron Solubility Limits versus Core Fluid Temperature



SNPB 2-3: Are there high concentrate boric acid storage tanks in the CR-3 NSSS? If so, please provide the boric acid concentration and volume of the tanks. Please also provide the maximum flow rate of the tanks into the reactor coolant system.

Response:

CR-3 has two concentrated BASTs each with a capacity of ~7000 gallons. The CR-3 BAP analysis conservatively uses a flow of 50 gpm for the first 45 minutes of the transient. This flow is not normally in-service; but, is being confirmed as closed in EOP revisions. A maximum boron concentration of 14,000 ppm was used for the BASTs.

SNPB 2-4: Please explain how boric acid precipitation is prevented for all small break loss-of-coolant accidents (SBLOCAs).

Response:

HL-BD is initiated prior to the time the core could reach the solubility limit for SBLOCA scenarios. This flow path causes forward flow though the core that dilutes the boron concentrations in the core region.

SNPB 2-5: Please provide the boric acid concentration vs time for the limiting large break loss-of-coolant (LBLOCA) and the limiting SBLOCA?

Response:

A spectrum of nine different break sizes has been considered and boron concentrations versus time and long-term RCS pressure response were calculated for each. With the assumption of no RVVV liquid overflow or hot leg nozzle gap flows, the largest LBLOCA with the minimum pressure will have the lowest core mixing volume and get to the solubility limit sooner. This case is described as the limiting LBLOCA case and its concentration is provided in Figure 4. It has an RCS pressure of 14.7 psia and core mixing volume near the time of peak core concentration of 722 ft^3. As break sizes decreases, the RCS pressure increases, as do the core mixing volume and the solubility limit. However, no credit is taken for the increase in the solubility limit versus saturation temperature, so the core concentration predictions during the transient are lower only because the core mixing volume increases with pressure. The imposed constant solubility limit used on smaller break sizes with core mixing volumes (that may be near or at a capped maximum value of 1200 ft^3) results in similar core concentrations for these SBLOCA cases. These considerations make it difficult to define the limiting SBLOCA. However, the RAI asks for the limiting SBLOCA so it is characterized as the case with an RCS pressure that will not use HLI to actively control the core boron concentration. The pressure for this case is an uncertainty adjusted minimum coinciding with an indicated 140 psia maximum pressure that HLI can be used to mitigate. The maximum pressure uncertainty is roughly 70 psid, so an RCS pressure could be anywhere between roughly 70 and 140 psia. An RCS pressure of 71 psia (just above the minimum uncertainty value) was selected because it has the smallest core mixing volume for cases in this pressure range. Figure 5 provides the core boron concentration for a 71 psia SBLOCA that credits operator action to open the HL-BD at 4.5 hours. A mixing volume of 1011 ft^3 was used with a conservatively low flow of 50 gpm of out of the hot leg injection blowdown line after 4.5 hours.

At least nine different cold leg pump discharge break sizes were considered in the spectrum of large and small LOCAs spanning long-term RCS pressures from 14.7 to 140 psia. Core boron concentrations were calculated for these cases with bounding core mixing volumes. The required operator action time is available from each case that needs initiation of the boric acid dilution flow prior to the time the core concentration reaches the imposed solubility limit at 212 F. The most limiting time is a key input into the emergency operator procedure guidance for CR-3 at the EPU power level.

Figure 4: LBLOCA Core Boric Acid Concentration versus Time.

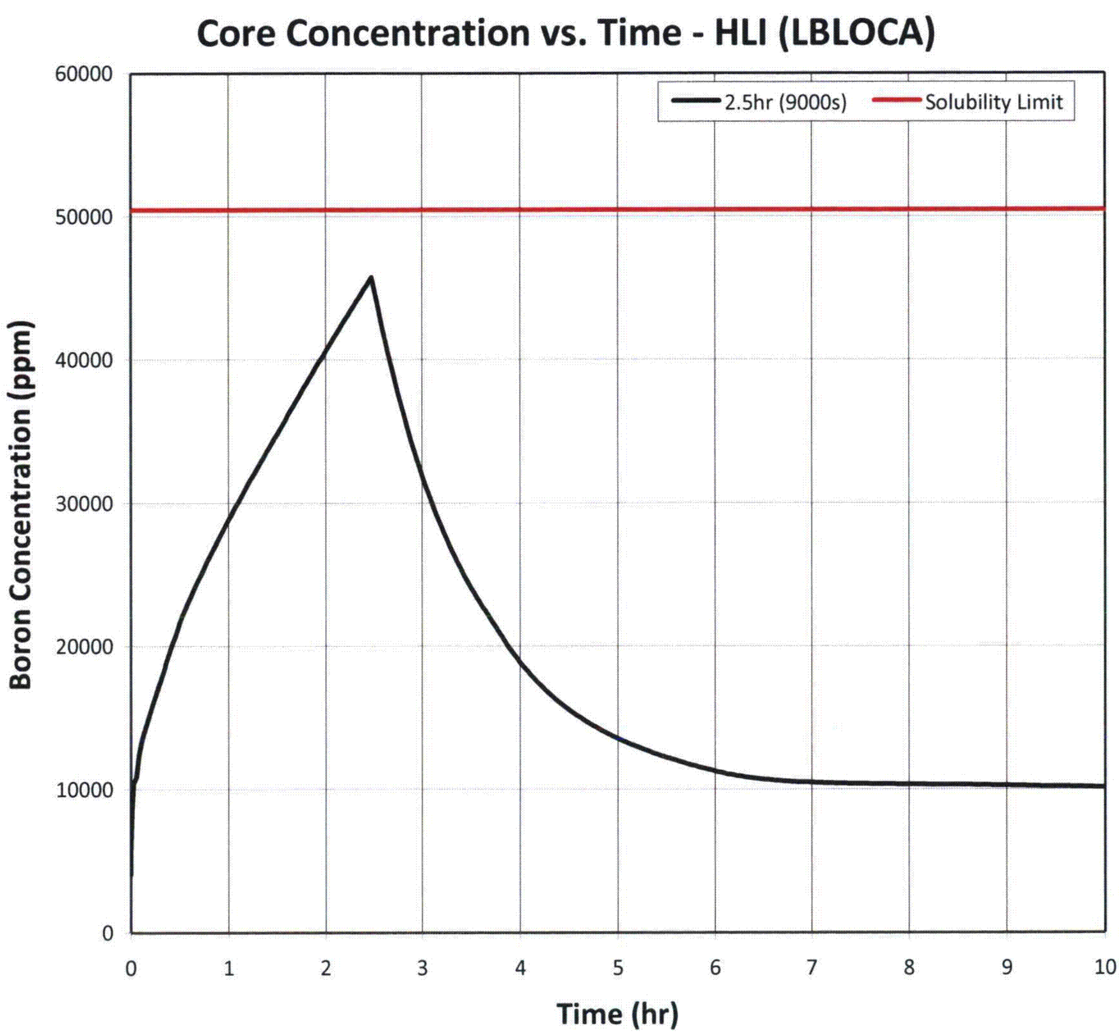
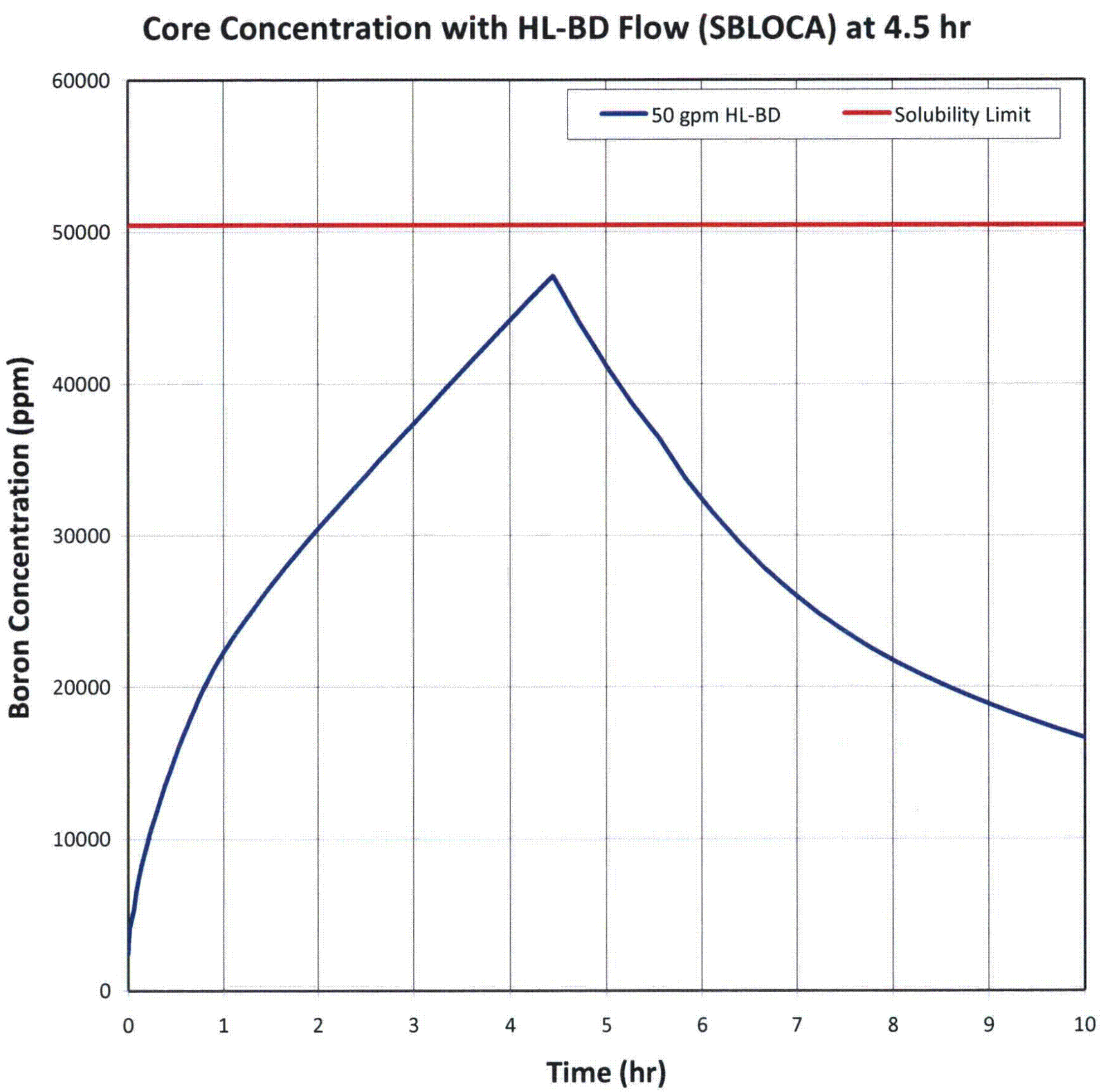


Figure 5: SBLOCA Core Boric Acid Concentration versus Time.



SNPB 2-6: Please provide the detailed results and write-up with accompanying plots, tables of key inputs/assumptions, and table of key event timings for the spectrum of breaks comprising the LBLOCA and SBLOCA spectrum analyses at EPU conditions.

Response:

This information was provided in the LOCA summary report for the CR-3 EPU that was provided to the NRC on July 17, 2012 via Reference 3.

SNPB 2-7: Please identify the limiting conditions assumed for boric acid precipitation calculations for the limiting LBLOCA and SBLOCA analyses.

Response:

The key parameters used in the CR-3 BAP analyses are provided in the following list.

- A 1.2 multiplier on the 1971 infinite operation fission product decay and a conservative actinide contribution is added.
- A maximum BWST usable liquid volume of 383,200 gallons with a boron concentration of 3,000 ppm.
- A high initial reactor coolant system boron concentration of 2400 ppm was used.
- The core flood tank initial boron concentration was 4000 ppm.
- It is assumed that the BAST is injecting liquid at a flow rate of 50 gpm with 14,000 ppm boron for up to 45 minutes during the event.
- The ECCS injection temperature during sump recirculation is modeled conservatively high at a constant 200 F value for core boiloff matchup and concentration purposes.
- A conservatively low 50,300 ppm water solubility limit based on the solubility limit at 212 F (This is equivalent to 24.2 w% H_3BO_3 including an imposed 4 w% reduction at 212 F).
- The lower plenum is not included in the core mixing volume.
- The liquid fluid volume below the minimum of either: the upper plenum mixture level or the bottom of the RVVV elevation is used to determine the core mixing volume.
- Voiding is considered in the core and upper plenum regions.
- The core mixing volumes are determined as a function of time post trip and pressure. A conservative bottom-peaked core power shape is used to minimize the core mixing volume.
- The limiting core concentration calculations do not include credit for any RVVV liquid overflow or hot leg nozzle gap flow.
- No boron carryout with the steam was credited in the core concentration calculations.
- Heat capacity of the core and passive metal in the refilled regions of RV are included as a heat source that can boil off core liquid.
- Sump mixed-mean boron concentration transport delays are used.
- The HLI flow is held constant at 400 gpm at pressures of 140 psia and below for the LBLOCA cases at low RCS pressures. Actual HLI flows are expected to be higher than 400 gpm, especially if two LPI pumps are in operation.
- The SBLOCA cases at higher RCS pressures use a HL-BD minimum flow of 50 gpm. This flow is conservatively low because it is less than the expected gravity drain flow rate.

SNPB 2-8: Since the steam flow rate due to core boil-off leaving the two-phase surface in the inner vessel during a LOCA contains boric acid, please demonstrate that boric acid crystals do not collect on the vent valves sufficiently to reduce this flow area or increase the resistance for this path to a break in the cold leg. What is the reduction in flow area assumed in the boric acid precipitation analysis?

Response:

The RVVVs are internal flapper valves that are hinged on the outside (downcomer side) and above the RVVV opening in the core barrel. The core steam flowing through the RVVVs does not directly impinge on the hinge mechanism. The hinge pin is contained within bushings mounted to the core barrel wall. The hinge pins can rotate on the inside of the bushings and the bushings can rotate inside the mountings. So the RVVVs are free to move from several degrees of freedom and they will not bind from boric acid carryover in the steam. Also, no plateout of boric acid will occur in or on the RVVVs, so no reduction of flow area or performance is modeled in the CR-3 BAP or LTCC analyses.

Boric acid is slightly volatile in steam based on the undissociated fraction of pure boric acid, the distribution coefficient, and the core boron concentration. Prior to the inclusion of sump pH additives, a small fraction of boron based on the mixture region boric acid concentration is carried out with the steam. However, introduction of the sump pH additives, which occurs passively as the trisodium phosphate in the sump is injected into the core following sump switchover, reduces the undissociated fraction of boric acid, greatly reducing the carryout of boron in the steam. With or without the carryout reductions, boron carryout with steam has been considered and it will not affect the RVVV performance.

Another potential binding mechanism that could affect RVVV performance is the condensation of the steam in the bushing region of the RVVVs. The RVVV mounting mechanism and its entire metal structure is going to be at the same temperature or slightly hotter than the steam flowing through the RVVVs due to the initial stored energy in the metal from steady-state operation. Therefore, there is no plausible mechanism for condensing on the bearings or plateout that could impair the RVVV freedom of motion. Even if it did collect on the valve bushings, it would have to occur when the RVVVs were open and not when they were closed.

The plateout of boron on the RVVV face or the hole in the core barrel wall will not occur. If plateout were postulated to occur here, it would also occur on control rod guide housing metal structures in the reactor vessel upper plenum and head above the mixture level. This plateout would be on the outside of the control rod guide column weldments, core outlet plenum cylinder first and then later on the RVVVs. If this plateout occurred, the boron concentration in the core would be lower.

Verbal Add-on Question: Describe the hole and slot sizes in the baffle region former plates and holes and slots in the baffle plates that can provide alternate flow paths if the core inlet plugged with debris.

Response:

The original draft RAIs included some questions on the potential core inlet flow losses due to debris accumulation at the core inlet. This resistance change from a debris plug can be important if a portion of the lower plenum was used in the core mixing volume or if it restricted flow from entering the core. The CR-3 BAP analyses do not credit any of the lower plenum volume in the core mixing volume. Core cooling flows can also enter the core through an alternate flow path via the core baffle region if the core inlet was postulated to plug with debris. This alternate flow path has large openings from the lower plenum into the bottom of the baffle region that will not plug with debris. The core baffle plates have holes and slots that connect the baffle to the core region. These openings are of sufficient size such that they will not be plugged by the small debris passing through the sump screens and entering the RCS. [

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Using the combined resistance of this alternate flow path, a pressure drop can be computed to provide the core cooling flow through this alternate flow path. First the decay heat is established at a very early time of 20 minutes post-trip following a LBLOCA. Sump recirculation has not started yet but could start soon after this time if both ECCS trains were in operation. If the lowest possible pressure of 14.7 psia is used, the most limiting flow rates can be computed.

First the decay heat is computed at 20 minutes post trip using 120% ANS 1971 infinite operation fission products plus actinides.

$$Q_{DH\ @\ 20\ min} = P_o * P/P_o |_{t = 20min} = 3026\ MWt * 0.0256 * 948\ Btu/s/MWt = 73400\ Btu/s$$

Next the required flow to match the decay heat at 20 minutes post trip at 14.7 psia is computed using the core inlet temperature is at a conservatively high value of 200 F.

$$W_{DH\ @\ 20\ min} = Q_{DH\ @\ 20\ min} / (h_g - h_{in}) = 73400\ \text{Btu/s} / (1150.5 - 168)\ \text{Btu/lbm} = 74.7\ \text{lbm/s}$$

The required decay heat flow is used with the combined resistance for the alternate flow path and the Bernoulli equation to calculate the form loss pressure drop for the alternate flow path.

$$\Delta P_{alt\ FP} = (k_{alt\ FP} / A^2_{alt\ FP}) * (W_{DH\ @\ 20\ min})^2 / (2\ \rho_{f\ in} * g_c)$$

[

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This form loss pressure drop is inconsequential so the alternate flow path can provide sufficient core cooling flows such that total core inlet plugging with debris is of no consequence.

During the LTCC portion of the transient, flow patterns change and the baffle flows are downward. However, if it is postulated that the core inlet was totally blocked by debris, the baffle flow would change to the upward flow and ECCS could still enter the core roughly 1/3 of the way from the bottom of the core. The net minimum resistance for this path is made up of the baffle inlet, two former plates, and the holes in the baffle plates. This flow path that has large holes that would not plug with debris so it remains an alternate flow path that makes the core inlet flow resistance change due to debris inconsequential from a core cooling perspective. Therefore, since the lower plenum is not credited in the core mixing volume boric acid precipitation analyses and a continuous core flow is provided via an alternate open flow path, there is no need to characterize the core inlet resistance change from a debris plug.

4.0 Summary

Responses were provided to the majority of the NRC questions transmitted in the August 2, 2012, Requests for Additional Information on the CR-3 EPU (ML12202A060). The requested information that was not provided was for questions that are not needed to perform boric acid precipitation for the CR-3 plant with RVVVs. An additional question was added to the transmitted list based on a phone call between Progress and the NRC that occurred after the RAIs were transmitted.

5.0 References

1. Letter from J. A. Franke (CR-3) to US NRC, Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate (ML11207A442), June 15, 2011.

2. Letter from S. P. Lingam (NRC) to J. A. Franke (CR-3), Crystal River Unit 3 Nuclear Generating Plant – Request for Additional Information for Extended power Uprate License Amendment Request (TAC No. ME 6527) (ML12202A060), August 2, 2012.

3. Letter from J. A. Franke (CR-3) to US NRC, Crystal River Unit 3 – Supplement to the Extended Power Uprate License Amendment Request #309 (TAC No. ME6527) (ML12205A356), July 17, 2012.

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ATTACHMENT D

SUPPLEMENT – SUMMARY OF THE REACTOR COOLANT SYSTEM HOT LEG BLOWDOWN LINE DESIGN

SUPPLEMENT – SUMMARY OF THE REACTOR COOLANT SYSTEM HOT LEG BLOWDOWN LINE DESIGN

BACKGROUND

Attachment 1 of the Crystal River Unit 3 (CR-3) Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 1) provides a request to approve crediting the new Reactor Coolant System (RCS) hot leg injection (HLI) method of precluding boron precipitation in the reactor vessel during design basis accidents at EPU conditions thereby eliminating the need for the active boron precipitation mitigation methodologies required in the current CR-3 Operating License and Section 4.2, "Boron Precipitation Mitigation," of Attachment 1 of the CR- 3 EPU LAR provides a discussion related to this request.

This supplement revises this request to include approval for crediting the new RCS Hot Leg Blowdown (HLBD) System as an additional method to prevent boron precipitation in the reactor vessel during design basis accidents at EPU conditions.

Existing procedural guidance requires that the boron precipitation mitigation function occur after a certain boron concentration in the core is reached. Similar to the RCS HLI method of precluding boron precipitation proposed in the EPU LAR, the RCS HLBD System will not be based on core boron concentration and precludes the need for continued use of the boronometer.

The RCS HLBD System is designed to provide an additional method of boron precipitation mitigation following a loss of coolant accident (LOCA) (See Figure 1). If, after a break in the RCS, pressure remains above 140 psia, the proposed RCS HLI flow-path may be ineffective since the RCS pressure exceeds the shut-off head of the low pressure injection pumps. Therefore, an alternative strategy to achieve adequate dilution flow is needed to mitigate boron precipitation for all break sizes.

Appendix D of the CR-3 EPU Technical Report (Reference 1, Attachment 5) contained a proprietary AREVA NP, Inc. technical document, 51-9161089-000, "Core Boric Acid Dilution Control for CR-3 at EPU Conditions," that provided a summary of the primary method of mitigating boron precipitation during a LOCA from EPU operating conditions. This method has been and remains the addition of an RCS HLI flow-path to be placed in service following breaks that rapidly and fully depressurize the RCS. In addition, discussions were included in the EPU Technical Report addressing small break LOCAs where the RCS remains pressurized. The discussion concluded that such breaks could be managed in a manner to preclude precipitation. Upon further review, it was determined that expanding the boron precipitation mitigation capabilities facilitates forward flow through the reactor core and aids RCS depressurization.

Manually initiating the RCS HLBD System during a small break LOCA creates a flow-path through the reactor core and out the HLBD line. Use of this option allows for rapid dilution and removal of boron which precludes boron precipitation.

While the RCS HLBD System is similar to the currently licensed dump-to-sump boron precipitation mitigation strategy, it is superior in a number of ways: 1) The RCS HLBD System can be placed in service with either or both Emergency Core Cooling System (ECCS) trains in operation; 2) the HLBD line discharge does not directly impact the emergency sump by discharging inside the containment primary shield wall and migrating to the emergency sump; 3) the HLBD System meets the single-failure criterion specified in CR-3 Final Safety Analysis

Report Criterion 1.4.38; and 4) the HLBD line is large enough to be effective at EPU operating conditions.

This change will enhance the ability of the ECCS to mitigate boron precipitation in the core region due to post accident coolant boil-off and will help to ensure the core geometry remains amenable to cooling as required by 10 CFR 50.46(b)(4).

SYSTEM DESCRIPTION

The HLBD line taps into the proposed RCS HLI line vertical piping approximately 7 feet below the RCS hot leg connection, between the upstream HLI pressure isolation valve and the Decay Heat System drop line (see Figures 2 and 3). The 2.5 inch HLBD line will span along the containment primary shield wall. At this point, the 2.5 inch pipe will split into two flow-paths; A train and B train; with each flow-path consisting of the following:

- a 2.5 inch, normally open, manual isolation valve for testing and maintenance of the downstream HLBD valves;
- a 2.5 inch-to-2 inch reducer located downstream of the manual isolation valve;
- two 0.75 inch, normally closed, manual test valves installed in series for testing the associated HLBD solenoid operated valves (SOVs); and
- two 2 inch, normally closed, direct current (DC) SOVs installed in series; the SOVs will be energize-to-open and will fail closed on loss of power.

The new HLBD System is designed to operate at an RCS maximum pressure and temperature of 2500 psi and 650°F, respectively; and provides an alternative flow-path to prevent boron precipitation in the reactor core with core conditions varying from atmospheric pressure at 70°F to the maximum design pressure and temperature.

The power supply and all related wiring for the HLBD SOVs will originate from independent and redundant safety related power sources; DC distribution panels DPDP-5A and DPDP-5B. Electrical components and wiring will be located above the maximum expected flood zone of 102-ft. elevation.

The HLBD SOV control switches and valve position indicators will be located in the 480V switchgear rooms which is part of the CR-3 control complex habitability envelope (see Figures 4 and 5) and is designated as a mild environment. An operator will be dispatched to close the associated circuit breakers and open the valves. Valve position indications at the local operating station will be used to indicate that the HLBD SOVs have changed position and the HLBD flow- path has been established.

Both of the HLBD flow-paths will discharge onto a floor plate inside of the containment primary shield wall to reduce the momentum of the water/steam discharging through the HLBD line. The location of the blowdown lines inside the containment primary shield wall provides for physical separation and a tortuous path to the emergency sump which minimizes jet impingement and debris generation and precludes interaction with sump performance.

Each HLBD line is designed to provide sufficient flow to preclude boron precipitation during design basis accidents from EPU conditions.

OPERATIONAL ASPECTS

General Conditions

In all operating modes, the HLBD SOV control switches will remain in the off position with the corresponding circuit breakers locked open. Closing the circuit breakers and moving the control switches to the open position will open the HLBD SOVs. Repositioning of the control switches to the closed position will close the HLBD SOVs. Re-opening of the circuit breakers ensures the HLBD SOVs cannot be accidentally opened, and de-energizes the HLBD System preventing a spurious opening of the HLBD SOVs. The HLBD System control switches and SOV position indicating lights are contained within the control complex habitability envelope. The HLBD SOVs will be tested in accordance with the plant Inservice Testing Program on a refueling outage frequency to assure continued Operability of the HLBD System. Additionally, the RCS HLBD System valves will be verified in the correct position and the HLBD SOV breakers verified locked open prior to plant startup following refueling.

Accident

During a LOCA, electrical power will be aligned to the HLBD SOVs and the opening of the HLBD SOVs will be directed by the main control room in accordance with approved procedures. Both aligning electrical power and opening the HLBD SOVs will be performed by an operator within the control complex habitability envelope. The operator will communicate to the main control room when these actions have been completed.

Refer to the Boron Precipitation Prevention Flow Chart (see Figure 6) for additional guidance on operation of RCS HLI and HLBD initiation for boron precipitation control. The following is a summary of the decisional logistics for determining the boron precipitation control method employed during a LOCA:

Path 1
Where adequate subcooling margin is maintained, no boron precipitation mitigation actions are required.

Path 2
If adequate subcooling margin does not exist and RCS pressure is below 140 psia, then RCS HLI will be established in accordance with approved procedures to achieve boron precipitation control.

The analyzed 140 psia (non error corrected value) corresponds to the RCS pressure that provides a boron dilution flow of 400 gpm for HLI. This dilution flow value exceeds core boil-off by at least 24 gpm to provide adequate dilution flow.

Path 3
During a small break LOCA where RCS pressure remains at or above 140 psia, the HLBD SOVs will be opened to provide adequate flow for boron dilution.

Path 4
If adequate subcooling margin does not exist and RCS pressure decreases to less than 140 psia with the HLBD SOVs open, further guidance may be provided by Technical Support Center to close the HLBD SOVs and open the HLI valves.

REGULATORY ASPECTS

The new RCS HLBD System design has been evaluated and it has been determined that this supplemental design does not alter the conclusion that the proposed EPU license amendment does not involve a Significant Hazards Consideration.

The HLBD System modification will not significantly increase the possibility of an accident previously evaluated as each HLBD line will have two HLBD SOVs arranged in series, the HLBD SOVs will be maintained closed, the operating power will be removed, and the associated circuit breakers locked open. Appropriate codes and standards, including American Society of Mechanical Engineers Boiler and Pressure Vessel Code requirements for procurement, installation, and testing of the new HLBD valves, will be used in the design of the HLBD System. Consequences related to the additional flow to the containment floor from the RCS HLBD System are bounded by the consequences resulting from a large break LOCA.

No new operating mode, or accident scenario was identified. Potential equipment failures of new RCS HLBD System have been considered and are equivalent or bounded by existing equipment failures or effects of these equipment failures are bounded by previously evaluated accidents. No new accidents or event precursors have been identified. Therefore, it is concluded that this plant modification will not create the possibility of a new or different kind of accident.

This plant modification does not involve a significant reduction in a margin of safety; rather, the ability to mitigate boron precipitation during a LOCA has been expanded such that all break sizes will be enveloped.

Reference

1. FPC to NRC letter dated June 15, 2011, "Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate." (ADAMS Accession No. ML112070659)

Low Pressure Injection Cross Tie & Boron Precipitation Mitigation System Layout

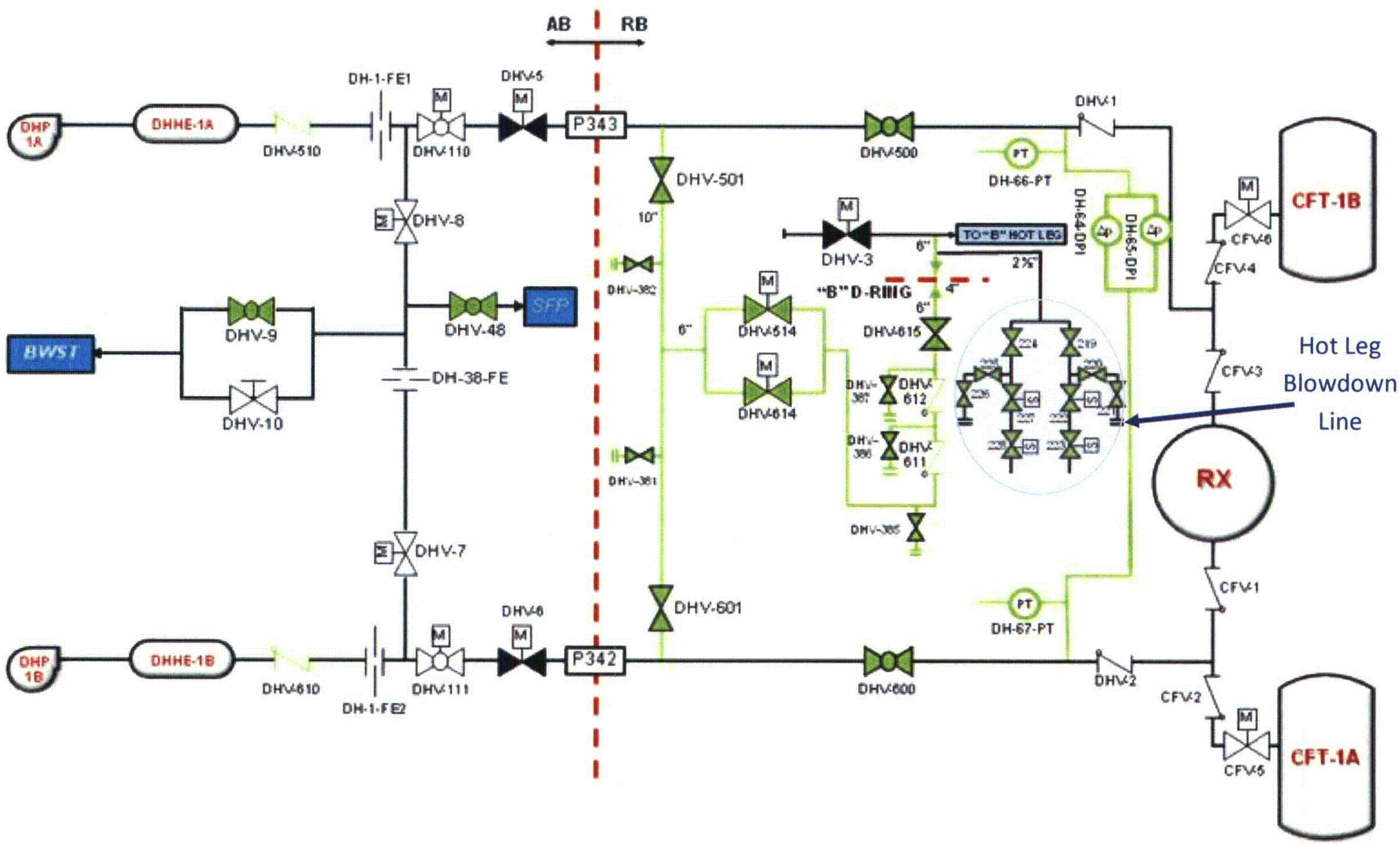

Figure 1

Low Pressure Injection Cross Tie & Hot Leg Blowdown Line Diagram

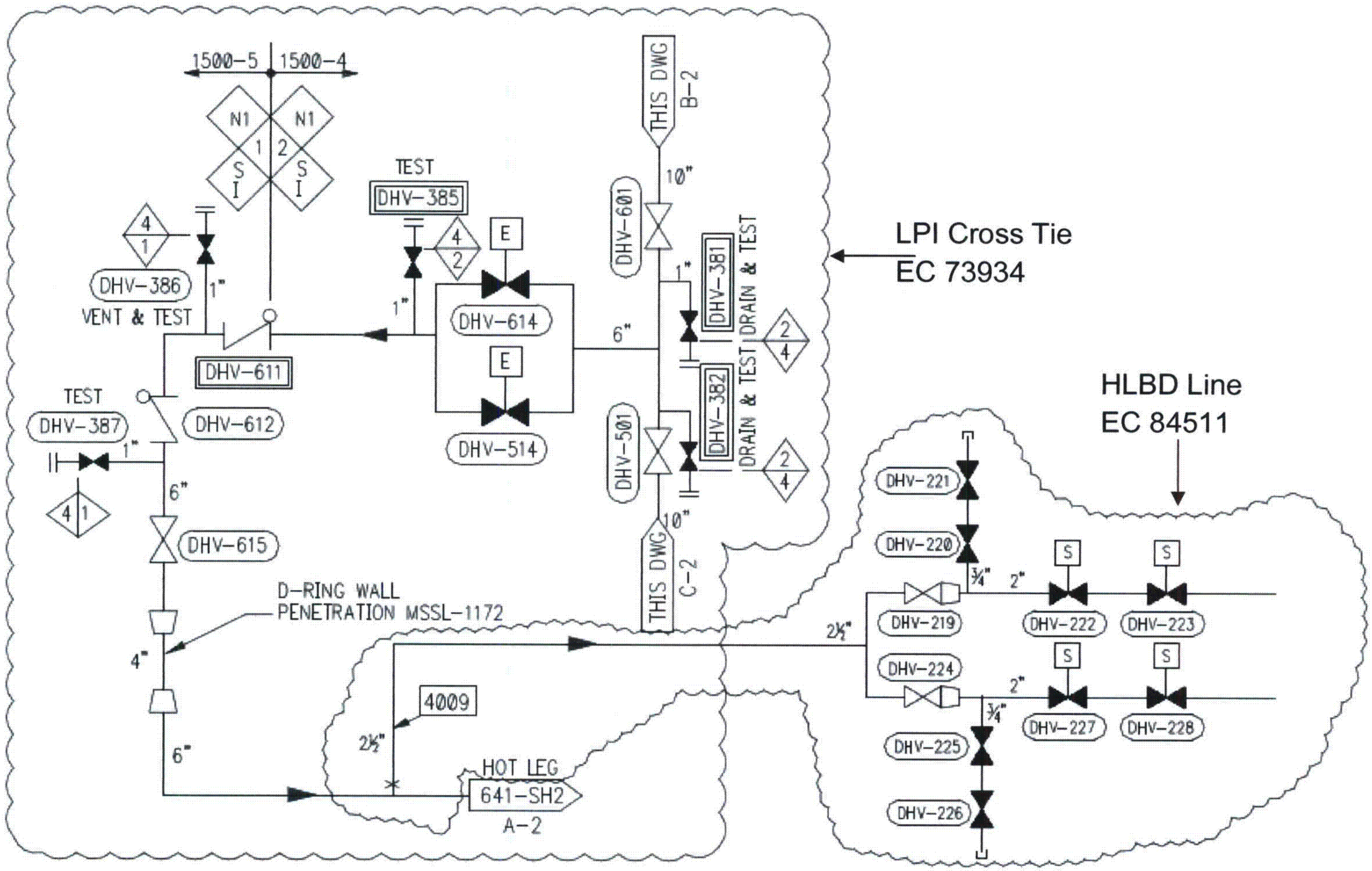

Figure 2

Hot Leg Blowdown Line - Isometric Drawing

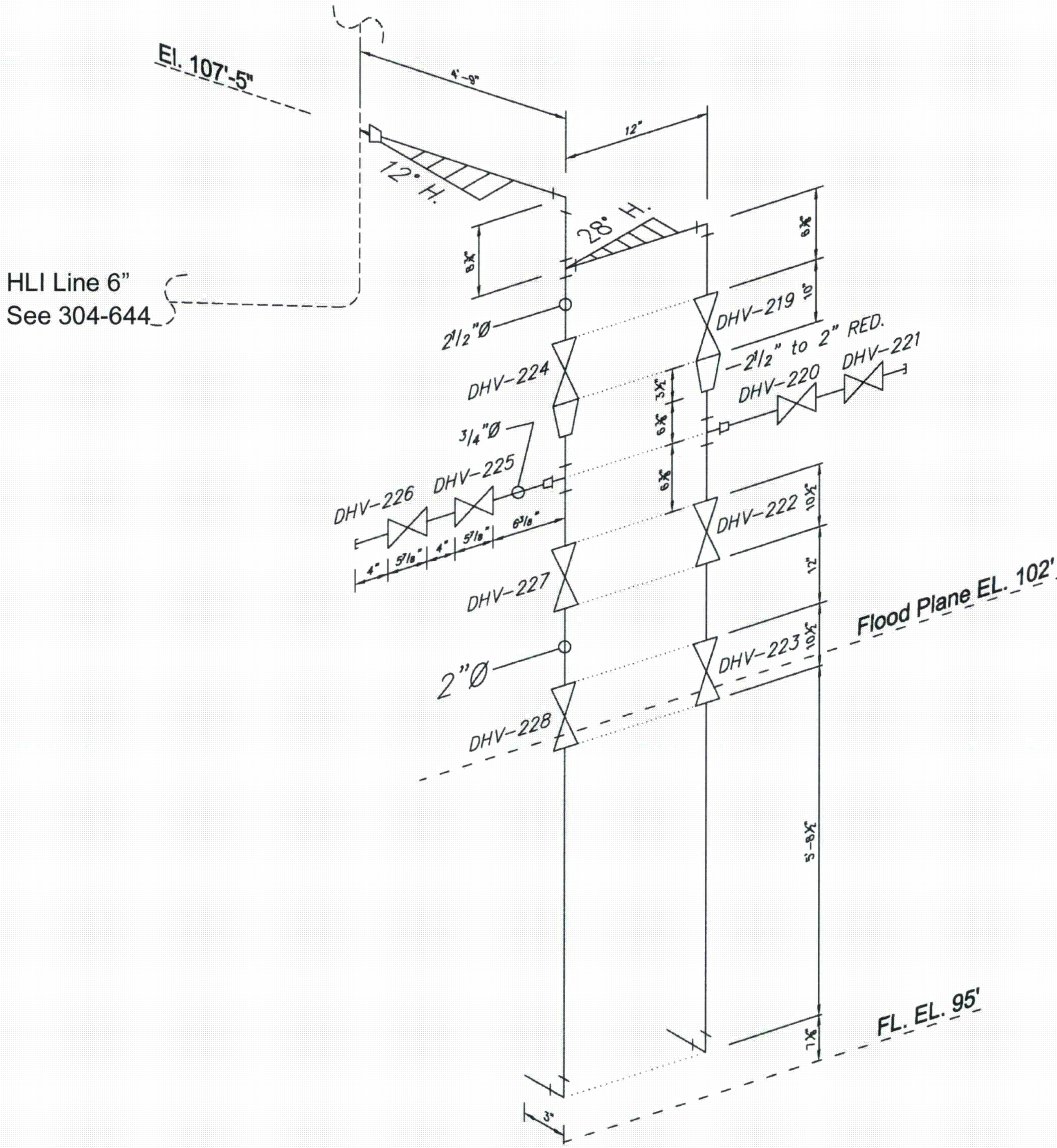

Figure 3

Proposed Switch Panel Layout for Hot Leg Blowdown

Train A

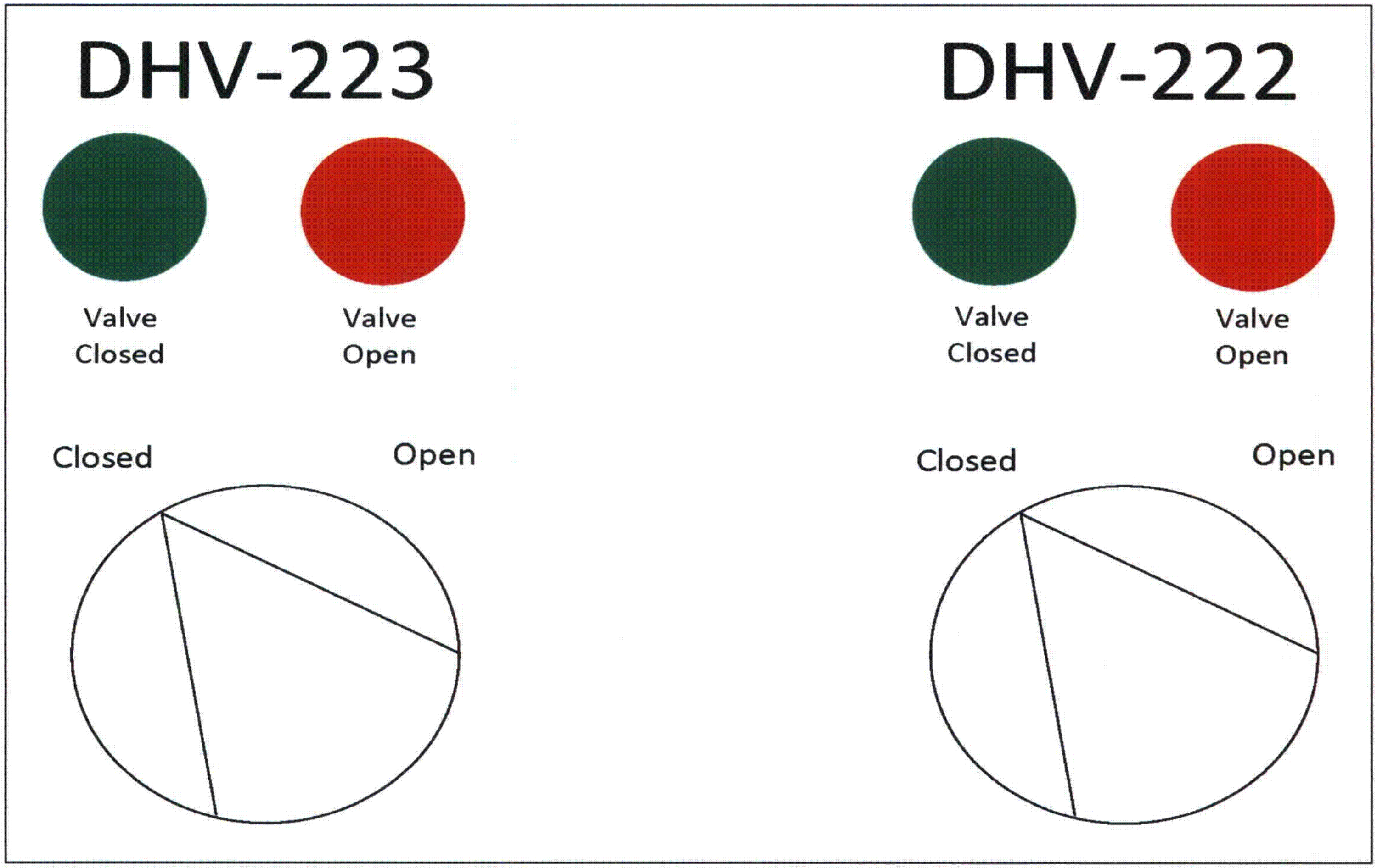

Figure 4

Control Panel Locations in Safety Related Switchgear Rooms
South-West Corner of the Control Complex – 124' Elevation

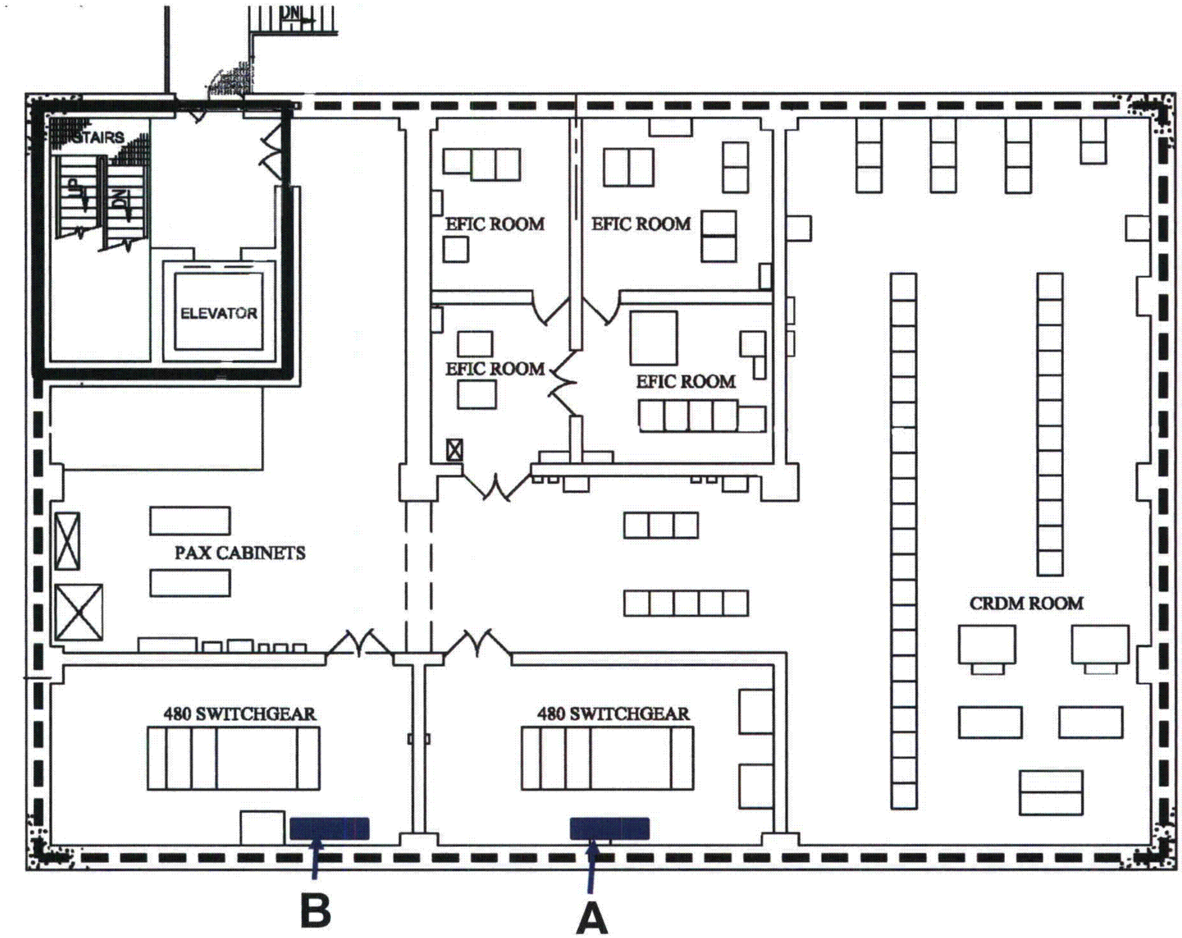

Figure 5

Boron Precipitation Prevention Flowchart

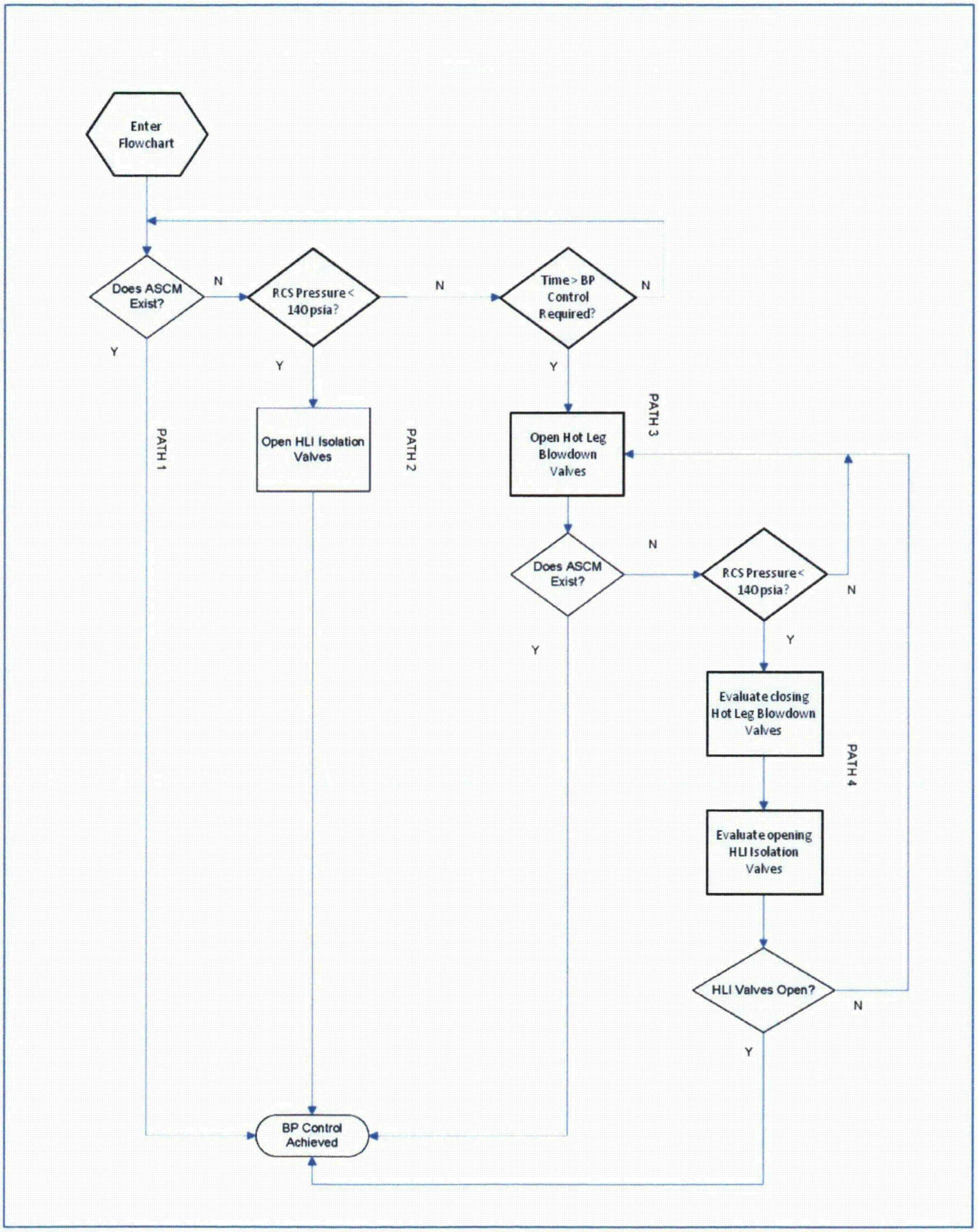

Figure 6

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ATTACHMENT E

MARKUP PAGES OF THE CR-3 EPU NO SIGNIFICANT HAZARDS CONSIDERATIONS EVALUATION

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power Corporation (FPC) has evaluated the proposed License Amendment Request (LAR) against the criteria of 10 CFR 50.92(c) to determine if any significant hazards consideration is involved. FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated?

The impacts of the proposed Extended Power Uprate (EPU) on plant systems, structures, and components (SSCs) were reviewed with respect to SSC design capability, and it was determined that following completion of plant changes to support the EPU, no SSC would exceed its design conditions or limits. Evaluations supporting those conclusions were performed and demonstrate that the equipment reliability and structural integrity will not be adversely affected by EPU. Control system studies demonstrated that plant response to operational transients under EPU conditions does not significantly increase reactor trip frequency, so there will be no significant increase in the frequency of SSC challenges caused by a reactor trip. The EPU does not create new failure modes for existing SSCs and eliminates the need for a single failure exemption currently in the Crystal River Unit 3 (CR-3) licensing basis for boron precipitation mitigation. A new potential inter-system loss-of-coolant accident (LOCA) mechanism is created by the installation of the HLI flow path, but the probability of an inter-system LOCA occurring has not significantly increased. Additionally, new pressure isolation valves installed in series between reactor coolant system (RCS) high and low pressure piping will minimize the likelihood of an inter-system LOCA. ASME Boiler and Pressure Vessel Code requirements for procurement, installation, and testing of the new pressure isolation valves will be followed. Also, a new potential steam line break mechanism is created by the inadvertent opening of both Atmospheric Dump Valves (ADVs) simultaneously. A new Inadequate Core Cooling Monitoring System (ICCMS) and Fast Cooldown System (FCS) are being installed which will automatically open both ADVs to support a small break LOCA in the event of a single failure in the Emergency Core Cooling System (ECCS). Modifications to ADVs and procurement, installation, testing and operation of the ICCMS and FCS will ensure that the potential inadvertent opening of both ADVs is minimized. Also, any adverse consequences of inadvertent opening both ADVs are bounded by the consequences of the main steam line break accident.

Also, a new potential LOCA flow path is created by the installation of the RCS hot leg blowdown (HLBD) system; however, the probability of a LOCA is not significantly increased by maintaining the blowdown valves closed and deactivated during plant operation.

and the new HLBD valves

The fission product barriers; fuel cladding, RCS pressure boundary, and the containment building, remain fully capable of performing their design functions. The spectrum of previously analyzed postulated accidents and transients was evaluated, and effects on the fuel, the RCS pressure boundary, and the containment were determined. Specific accident scenarios (small break LOCA, locked reactor coolant pump rotor, and rod ejection accident) have been determined to potentially cause cladding rupture under EPU conditions in limited amounts, but the quantity of the failures and the consequences are bounded by the large break LOCA analysis. The fuel remains within the acceptance criteria of 10 CFR 50.46, the RCS pressure will not increase for normal operation as a result of EPU, and accident conditions remain within the ASME Boiler and Pressure Vessel Code limits as well as SSC design limits. Analysis has also confirmed that during the worst case accident (large break LOCA), the containment building remains within its design limit. These analyses were performed and demonstrate that existing RCS pressure boundary and containment limits are met and the effects on the fuel are such that dose consequences meet existing criteria at EPU conditions.

With the exception of the steam generator tube rupture (SGTR) accident, the EPU analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183, "Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors." The SGTR analyses were performed using current licensing basis methodologies. Safety margins have been evaluated and margin has been retained to ensure that the analyses adequately bound the postulated limiting event scenarios. These analyses indicate increased doses for certain analyzed accidents. Various factors contribute to these increases. Several actions have been taken to limit the increased consequences. Installation of the new RCS HLBD system and modifications ~~Modifications~~ to the Low Pressure Injection (LPI) and ADV systems are being made to ensure that the consequences of previously evaluated accidents are not significantly increased. Specifically, these modifications will enhance the plant response capabilities to a small break LOCA and improve the method for boron precipitation mitigation. The proposed amendment reduces the maximum allowed RCS specific activity. The limits on specific activity ensure that the doses remain within the regulatory limits during analyzed transients and accidents. The maximum allowed operating containment pressure is being reduced from 3.0 psig to 1.5 psig; thereby ensuring that maximum peak containment internal pressure does not exceed limits in the event of a design basis accident.

The revised accident analyses demonstrate that the plant site and the dose-mitigating Engineered Safety Features remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR Part 50.67 "Alternative source term,". Therefore, the consequences of analyzed accidents are not significantly impacted by the proposed EPU.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) *Does not create the possibility of a new or different kind of accident from any accident previously evaluated?*

Equipment that could be affected by EPU has been evaluated. No new operating mode, or accident scenario was identified. The LPI, HLI, and ADV systems will be modified and an ICCMS and FCS will be installed to better respond to accident and non-accident conditions. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. The limiting accident remains the large break LOCA and analysis results are acceptable under EPU conditions. EPU uses developed technology and applies it within capabilities of existing or modified plant safety-related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). Modifications to existing SSCs and installation of new SSCs are designed in accordance with regulatory criteria to minimize equipment failures. Potential equipment failures of new or modified SSCs have been evaluated and postulated failures are equivalent or bounded by existing equipment failures or effects of these equipment failures are bounded by previously evaluated accidents. No new accidents or event precursors have been identified.

The Technical Specification (TS) revisions required to implement EPU continue to assure that the plant is operated within the limits established for safe operation of the plant. Additionally, the limits in the TS reflect the initial conditions for the safety analyses performed to demonstrate the plant can mitigate the effects of accidents and ensure public safety by maintaining offsite doses within the limits in 10 CFR 50.67. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated.

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

(3) *Does not involve a significant reduction in a margin of safety?*

Structural evaluations performed at EPU conditions demonstrated that calculated loads on affected SSCs after modification, if necessary, remain within their design allowable for all design basis event categories. ASME Code requirements continue to be met.

Fuel performance evaluations were performed using parameter values appropriate for a reload core operating at EPU conditions. Those evaluations demonstrate that fuel performance acceptance criteria continue to be met. Core reload evaluation processes ensure that the planned fuel load in the first reactor core to be operated at the increased power level, will meet applicable regulatory criteria.

LOCA and non-LOCA safety analyses were performed under EPU conditions. ECCS performance was shown to meet the criteria of 10 CFR 50.46. Small break LOCA, locked rotor, and rod ejection scenarios may, under EPU conditions, have some potential of resulting in a limited amount of fuel cladding failure, but the analyses conclude that the plant remains within the acceptance criteria of 10 CFR 50.46. Large break LOCA scenarios also satisfy the criteria of 10 CFR 50.46 under EPU conditions, and the large break LOCA remains the most limiting accident at EPU conditions. LOCA analyses indicate a small reduction in margin (< 5%) related to maximum local clad oxidation and hydrogen generation at EPU conditions and a slight improvement in peak clad temperature margin based on minor changes in LOCA design inputs and modifications that will enhance the plant response capabilities to LOCAs and improve the method for boron precipitation mitigation. The non-LOCA events identified in the CR-3 Final Safety Analysis Report, with the exception of the feedwater line break (FWLB) and rod ejection accidents, were shown to meet existing acceptance criteria. The FWLB and rod ejection accidents were shown to meet the acceptance criteria specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The containment building response to mass and energy releases was evaluated under EPU conditions. The evaluations indicated that temperature and pressure limits were met.

No plant changes associated with the EPU reduce the degree of component or system redundancy. The small break LOCA response will require two High Pressure Injection (HPI) pumps injecting. In the event of inadequate HPI and loss sub-cooling margin (SCM), secondary depressurization will be achieved via a new ICCMS and FCS, which includes automatic actuation of the ADVs, thereby assuring that the reactor core receives the necessary ECCS flow to minimize core damage and satisfy the requirements of 10 CFR 50.46. The features required to automatically open ADVs during a small break LOCA are incorporated into proposed TS 3.3.19, “Inadequate Core Cooling Monitoring System (ICCMS) Instrumentation,” and TS 3.3.20, “Inadequate Core Cooling Monitoring System (ICCMS),” and TS 3.7.20, “Fast Cooldown System (FCS).” The ICCMS and FCS are configured and supported such that a single failure will not prevent completion of the ECCS safety function.

To support this enhancement, operators will be provided indication of subcooling margin and HPI System flow adequacy to ensure actuation of the FCS. A new safety related display system will be available to determine when insufficient subcooling margin is available and HPI System flow is inadequate. Additional modifications will, upon indication of loss of subcooling margin, automatically trip the RCPs and raise steam generator secondary side level to the inadequate subcooling margin level. These automatic actions replace the current actions performed by the operator, thus reducing the reliance on manual operator action for event mitigation.

Operator training programs will be revised in accordance with the industry standard systematic approach to training process and appropriate training will be provided on all plant modifications, administrative/technical requirement changes, Technical Specification revisions, and procedure revisions. The CR-3 simulator will be updated and tested in sufficient time to provide effective reinforcement of procedure and plant physical changes as well as build proficiency with required manual operator actions.

Based on the above, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, CR-3 concludes that the proposed LAR presents a 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.

6.0 ENVIRONMENTAL IMPACT EVALUATION

The environmental considerations evaluation is contained in Attachment 9, "Supplemental Environmental Report Extended Power Uprate." It concludes that the EPU will not result in a significant change in non-radiological impacts on land use, water use, waste discharges, terrestrial and aquatic biota, transmission facilities, or social and economic factors, and will have no non-radiological impacts other than those evaluated in the Supplemental Environmental Report. The Supplemental Environmental Report further concludes that the EPU will not introduce any new radiological release pathways, will not result in a significant increase in occupational or public radiation exposures, and will not result in significant additional fuel cycle environmental impacts.

Therefore, the proposed amendment does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite nor does it involve a significant increase in individual or cumulative occupational radiation exposure.

7.0 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

CR-3 has determined that the proposed changes do not require any exemptions or relief from regulatory requirements and do not adversely affect conformance with any regulatory requirements differently than described in the CR-3 Final Safety Analysis Report. The exemption to 10 CFR 50 Appendix K, Item I.D.1, requirements for single failure considerations is being deleted due to a modification to the plant that will adequately provide the boron precipitation mitigation function and is designed to remain functional even with a single failure condition.

This LAR will not reduce the effectiveness of the safety related systems, structures, or components and will not require the plant to operate outside of analyzed limits. Therefore, 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) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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ATTACHMENT F

REVISED PAGES OF THE CR-3 EPU NO SIGNIFICANT HAZARDS CONSIDERATIONS EVALUATION

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power Corporation (FPC) has evaluated the proposed License Amendment Request (LAR) against the criteria of 10 CFR 50.92(c) to determine if any significant hazards consideration is involved. FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated?

The impacts of the proposed Extended Power Uprate (EPU) on plant systems, structures, and components (SSCs) were reviewed with respect to SSC design capability, and it was determined that following completion of plant changes to support the EPU, no SSC would exceed its design conditions or limits. Evaluations supporting those conclusions were performed and demonstrate that the equipment reliability and structural integrity will not be adversely affected by EPU. Control system studies demonstrated that plant response to operational transients under EPU conditions does not significantly increase reactor trip frequency, so there will be no significant increase in the frequency of SSC challenges caused by a reactor trip. The EPU does not create new failure modes for existing SSCs and eliminates the need for a single failure exemption currently in the Crystal River Unit 3 (CR-3) licensing basis for boron precipitation mitigation. A new potential inter-system loss-of-coolant accident (LOCA) mechanism is created by the installation of the HLI flow path, but the probability of an inter-system LOCA occurring has not significantly increased. Additionally, new pressure isolation valves installed in series between reactor coolant system (RCS) high and low pressure piping will minimize the likelihood of an inter-system LOCA. Also, a new potential LOCA flow path is created by the installation of the RCS hot leg blowdown (HLBD) system; however, the probability of a LOCA is not significantly increased by maintaining the blowdown valves closed and deactivated during plant operation. ASME Boiler and Pressure Vessel Code requirements for procurement, installation, and testing of the new pressure isolation valves and the new HLBD valves will be followed. Also, a new potential steam line break mechanism is created by the inadvertent opening of both Atmospheric Dump Valves (ADVs) simultaneously. A new Inadequate Core Cooling Monitoring System (ICCMS) and Fast Cooldown System (FCS) are being installed which will automatically open both ADVs to support a small break LOCA in the event of a single failure in the Emergency Core Cooling System (ECCS). Modifications to ADVs and procurement, installation, testing and operation of the ICCMS and FCS will ensure that the potential inadvertent opening of both ADVs is minimized. Also, any adverse consequences of inadvertent opening both ADVs are bounded by the consequences of the main steam line break accident.

The fission product barriers; fuel cladding, RCS pressure boundary, and the containment building, remain fully capable of performing their design functions. The spectrum of previously analyzed postulated accidents and transients was evaluated, and effects on the fuel, the RCS pressure boundary, and the containment were determined. Specific accident scenarios (small break LOCA, locked reactor coolant pump rotor, and rod ejection accident) have been determined to potentially cause cladding rupture under EPU conditions in limited amounts, but the quantity of the failures and the consequences are bounded by the large break LOCA analysis. The fuel remains within the acceptance criteria of 10 CFR 50.46, the RCS pressure will not increase for normal operation as a result of EPU, and accident conditions remain within the ASME Boiler and Pressure Vessel Code limits as well as SSC design limits. Analysis has also confirmed that during the worst case accident (large break LOCA), the containment building remains within its design limit. These analyses were performed and demonstrate that existing RCS pressure boundary and containment limits are met and the effects on the fuel are such that dose consequences meet existing criteria at EPU conditions.

With the exception of the steam generator tube rupture (SGTR) accident, the EPU analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183, "Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors." The SGTR analyses were performed using current licensing basis methodologies. Safety margins have been evaluated and margin has been retained to ensure that the analyses adequately bound the postulated limiting event scenarios. These analyses indicate increased doses for certain analyzed accidents. Various factors contribute to these increases. Several actions have been taken to limit the increased consequences. Installation of the new RCS HLBD system and modifications to the Low Pressure Injection (LPI) and ADV systems are being made to ensure that the consequences of previously evaluated accidents are not significantly increased. Specifically, these modifications will enhance the plant response capabilities to a small break LOCA and improve the method for boron precipitation mitigation. The proposed amendment reduces the maximum allowed RCS specific activity. The limits on specific activity ensure that the doses remain within the regulatory limits during analyzed transients and accidents. The maximum allowed operating containment pressure is being reduced from 3.0 psig to 1.5 psig; thereby ensuring that maximum peak containment internal pressure does not exceed limits in the event of a design basis accident.

The revised accident analyses demonstrate that the plant site and the dose-mitigating Engineered Safety Features remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR Part 50.67 "Alternative source term,". Therefore, the consequences of analyzed accidents are not significantly impacted by the proposed EPU.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) *Does not create the possibility of a new or different kind of accident from any accident previously evaluated?*

Equipment that could be affected by EPU has been evaluated. No new operating mode, or accident scenario was identified. The LPI, HLI, and ADV systems will be modified and an ICCMS and FCS will be installed to better respond to accident and non-accident conditions. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. The limiting accident remains the large break LOCA and analysis results are acceptable under EPU conditions. EPU uses developed technology and applies it within capabilities of existing or modified plant safety-related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). Modifications to existing SSCs and installation of new SSCs are designed in accordance with regulatory criteria to minimize equipment failures. Potential equipment failures of new or modified SSCs have been evaluated and postulated failures are equivalent or bounded by existing equipment failures or effects of these equipment failures are bounded by previously evaluated accidents. No new accidents or event precursors have been identified.

The Technical Specification (TS) revisions required to implement EPU continue to assure that the plant is operated within the limits established for safe operation of the plant. Additionally, the limits in the TS reflect the initial conditions for the safety analyses performed to demonstrate the plant can mitigate the effects of accidents and ensure public safety by maintaining offsite doses within the limits in 10 CFR 50.67. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated.

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

(3) *Does not involve a significant reduction in a margin of safety?*

Structural evaluations performed at EPU conditions demonstrated that calculated loads on affected SSCs after modification, if necessary, remain within their design allowable for all design basis event categories. ASME Code requirements continue to be met.

Fuel performance evaluations were performed using parameter values appropriate for a reload core operating at EPU conditions. Those evaluations demonstrate that fuel performance acceptance criteria continue to be met. Core reload evaluation processes ensure that the planned fuel load in the first reactor core to be operated at the increased power level, will meet applicable regulatory criteria.

LOCA and non-LOCA safety analyses were performed under EPU conditions. ECCS performance was shown to meet the criteria of 10 CFR 50.46. Small break LOCA, locked rotor, and rod ejection scenarios may, under EPU conditions, have some potential of resulting in a limited amount of fuel cladding failure, but the analyses conclude that the plant remains within the acceptance criteria of 10 CFR 50.46. Large break LOCA scenarios also satisfy the criteria of 10 CFR 50.46 under EPU conditions, and the large break LOCA remains the most limiting accident at EPU conditions. LOCA analyses indicate a small reduction in margin (< 5%) related to maximum local clad oxidation and hydrogen generation at EPU conditions and a slight improvement in peak clad temperature margin based on minor changes in LOCA design inputs and modifications that will enhance the plant response capabilities to LOCAs and improve the method for boron precipitation mitigation. The non-LOCA events identified in the CR-3 Final Safety Analysis Report, with the exception of the feedwater line break (FWLB) and rod ejection accidents, were shown to meet existing acceptance criteria. The FWLB and rod ejection accidents were shown to meet the acceptance criteria specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The containment building response to mass and energy releases was evaluated under EPU conditions. The evaluations indicated that temperature and pressure limits were met.

No plant changes associated with the EPU reduce the degree of component or system redundancy. The small break LOCA response will require two High Pressure Injection (HPI) pumps injecting. In the event of inadequate HPI and loss sub-cooling margin (SCM), secondary depressurization will be achieved via a new ICCMS and FCS, which includes automatic actuation of the ADVs, thereby assuring that the reactor core receives the necessary ECCS flow to minimize core damage and satisfy the requirements of 10 CFR 50.46. The features required to automatically open ADVs during a small break LOCA are incorporated into proposed TS 3.3.19, "Inadequate Core Cooling Monitoring System (ICCMS) Instrumentation," and TS 3.3.20, "Inadequate Core Cooling Monitoring System (ICCMS)," and TS 3.7.20, "Fast Cooldown System (FCS)." The ICCMS and FCS are configured and supported such that a single failure will not prevent completion of the ECCS safety function.

To support this enhancement, operators will be provided indication of subcooling margin and HPI System flow adequacy to ensure actuation of the FCS. A new safety related display system will be available to determine when insufficient subcooling margin is available and HPI System flow is inadequate. Additional modifications will, upon indication of loss of subcooling margin, automatically trip the RCPs and raise steam generator secondary side level to the inadequate subcooling margin level. These automatic actions replace the current actions performed by the operator, thus reducing the reliance on manual operator action for event mitigation.

Operator training programs will be revised in accordance with the industry standard systematic approach to training process and appropriate training will be provided on all plant modifications, administrative/technical requirement changes, Technical Specification revisions, and procedure revisions. The CR-3 simulator will be updated and tested in sufficient time to provide effective reinforcement of procedure and plant physical changes as well as build proficiency with required manual operator actions.

Based on the above, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, CR-3 concludes that the proposed LAR presents a 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.

6.0 ENVIRONMENTAL IMPACT EVALUATION

The environmental considerations evaluation is contained in Attachment 9, "Supplemental Environmental Report Extended Power Uprate." It concludes that the EPU will not result in a significant change in non-radiological impacts on land use, water use, waste discharges, terrestrial and aquatic biota, transmission facilities, or social and economic factors, and will have no non-radiological impacts other than those evaluated in the Supplemental Environmental Report. The Supplemental Environmental Report further concludes that the EPU will not introduce any new radiological release pathways, will not result in a significant increase in occupational or public radiation exposures, and will not result in significant additional fuel cycle environmental impacts.

Therefore, the proposed amendment does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite nor does it involve a significant increase in individual or cumulative occupational radiation exposure.

7.0 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

CR-3 has determined that the proposed changes do not require any exemptions or relief from regulatory requirements and do not adversely affect conformance with any regulatory requirements differently than described in the CR-3 Final Safety Analysis Report. The exemption to 10 CFR 50 Appendix K, Item I.D.1, requirements for single failure considerations is being deleted due to a modification to the plant that will adequately provide the boron precipitation mitigation function and is designed to remain functional even with a single failure condition.

This LAR will not reduce the effectiveness of the safety related systems, structures, or components and will not require the plant to operate outside of analyzed limits. Therefore, 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) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.