Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

April 13, 2007
3F0407-12

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

Subject: Crystal River Unit 3 – License Amendment Request #295, Revision 0, Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

References:
1. Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-430, Revision 2 (BWOG-104, Revision 1) dated December 22, 2003, "AOT Extension to 7 Days for LPI and Containment Spray (BAW-2295-A, Revision 1)"
2. Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-439-A, Revision 2 (WOG-165, Revision 0) dated December 19, 2005, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO"

Dear Sir:

In accordance with the provisions of 10 CFR 50.90, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., hereby submits License Amendment Request #295, Revision 0. The proposed amendment would revise the Crystal River Unit 3 (CR-3) Improved Technical Specification (ITS) as follows:

- Extend the allowed outage time (AOT) to seven days for one train of Reactor Building Spray inoperable, one train of Decay Heat Closed Cycle Cooling Water System inoperable, and one train of Decay Heat Seawater System inoperable,
- Add a new Condition for one Low Pressure Injection subsystem inoperable with an AOT of seven days,
- Add a new Condition for one Reactor Building Spray train inoperable coincident with one Containment Cooling train inoperable with an AOT of 72 hours,
- Eliminate second Completion Times from the CR-3 ITS, and
- Include editorial/administrative changes to provide clarity or delete obsolete information from the CR-3 ITS, Bases and Operating License.

FPC has evaluated the proposed license amendment request using both deterministic and probabilistic methodologies. These evaluations have determined that there are compensatory actions that can be taken during extended maintenance that can reduce overall risk. These

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Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

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actions are identified in Attachment F as regulatory commitments.

This submittal is consistent with the NRC-approved revisions of References 1 and 2.

Attachment A provides a description of the proposed change and confirmation of applicability. Attachment B provides existing pages from the CR-3 Operating License with proposed changes. Attachment C provides existing pages from the CR-3 ITS and Bases marked-up to show the proposed changes, and Attachment D shows these same changes presented more formally with revision bars. Attachment E documents a risk assessment performed by Progress Energy of the extended allowed outage times. Finally, Attachment F lists the regulatory commitments associated with the proposed change.

FPC requests approval of the proposed License amendment by September 30, 2007, with the amendment to be implemented within thirty days of issuance. Issuance by this date will permit online maintenance of the Decay Heat System prior to the upcoming outage and reduce dose.

In accordance with 10 CFR 50.91, a copy of this application with enclosures is being provided to the designated Florida State Official.

The CR-3 Plant Nuclear Safety Committee has reviewed this request and recommended it for approval.

If you have any questions regarding this submittal, please contact Mr. Paul Infanger, Supervisor, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/dar

Attachments:
- A. Description and Assessment
- B. Proposed Crystal River Unit 3 Operating License Changes
- C. Proposed Improved Technical Specification and Bases Changes (Mark-up)
- D. Proposed Improved Technical Specification and Bases Changes (Revision Bar Format)
- E. Low Pressure Injection and Reactor Building Spray AOT Extension Risk Assessment
- F. List of Regulatory Commitments

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector
State Contact

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young
Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 13th day of April, 2007, by Dale E. Young.

Ellen Deppolder
Signature of Notary Public
State of Florida

ELLEN DEPPOLDER
MY COMMISSION # DD 408539
EXPIRES: July 8, 2009
Bonded Thru Notary Public Underwriters

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

ATTACHMENT A

Description and Assessment

1.0 DESCRIPTION

The License Amendment Request (LAR) proposes to revise several pages of the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS). The proposed changes fall into one of three categories:

- Proposed changes made to support the extension of some allowed outage times (AOT) in the CR-3 ITS as presented in Technical Specification Task Force (TSTF) 430, Revision 2 (Reference 8.1),
- Proposed changes made to delete second Completion Times from the CR-3 ITS as presented in TSTF 439-A, Revision 2 (Reference 8.2), or
- Proposed editorial/administrative changes that either provide clarity or delete obsolete information from the CR-3 ITS, Bases and Operating License.

Each category of changes is presented and discussed separately.

Changes Proposed in Accordance with TSTF-430, Revision 2

TSTF 430 implements changes presented in Revision 1 to Topical Report BAW-2295-A (Reference 8.3) that extend the AOT for three specifications from 72 hours to seven days and introduces two new Conditions. Nuclear Regulatory Commission (NRC) approval of this topical report and TSTF was granted on July 15, 1999, and August 5, 2004, respectively. The changes are proposed for the following CR-3 ITS systems consistent with these documents:

- ITS 3.5.2, Low Pressure Injection (LPI) in Emergency Core Cooling Systems (ECCS) – Operating,
- ITS 3.6.6, Reactor Building Spray (BS) and Containment Cooling Systems,
- ITS 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System, and
- ITS 3.7.10, Decay Heat Seawater (RW) System.

Proposed changes are also included for the CR-3 Bases to reflect the increased AOTs.

Changes Proposed in Accordance with TSTF-439-A, Revision 2

TSTF 439-A (Reference 8.2) proposes to delete second Completion Times. In some CR-3 ITS, a second Completion Time exists for some Required Actions to establish a limit on the maximum time allowed for any combination of Conditions that result in a single continuous failure to meet the Limiting Condition of Operation (LCO). Typically, these second Completion Times are joined by an "AND" logical connector to the Condition-specific Completion Time and are written as "X days from discovery of failure to meet the LCO" (where "X" varies by specification). TSTF 439-A deletes these second Completion Times from the affected Required Actions. It also revises the discussion found in ITS 1.3, Completion Times, specifically Example 1.3-3, to state that alternating between Actions to operate indefinitely without satisfying the LCO is inappropriate and inconsistent with the basis for Completion Times.

Eliminating these second Completion Times is acceptable because these events are already controlled administratively by programs implemented to meet requirements of the Maintenance Rule. These controls ensure that the time allowed for any combination of Conditions that result in a single contiguous failure to meet the LCO is not improperly extended.

NRC approval of TSTF-439-A, Revision 2, was granted in Reference 8.5 which states the proposed change was incorporated into Revision 3.1 of the Standard Technical Specifications. Changes are proposed for the following CR-3 ITS consistent with the NRC approved TSTF-439-A:

- ITS 1.3, Example 1.3-3, Completion Times,
- ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems,
- ITS 3.7.5, Emergency Feedwater (EFW) System,
- ITS 3.8.1, AC Sources – Operating, and
- ITS 3.8.9, Distribution Systems – Operating.

Proposed changes are also included for the Bases and references for these ITS to reflect the removal of the second Completion Time.

Editorial/Administrative Changes

Some proposed editorial/administrative changes are also included in this LAR. One change removes remnants of one-time only changes from some specifications and from the license that are now obsolete. These were implemented as footnotes to some Completion Times in the following:

- ITS 3.5.2, ECCS – Operating,
- ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems,
- ITS 3.7.5, Emergency Feedwater (EFW) System,
- ITS 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System, and
- ITS 3.7.10, Decay Heat Seawater System.

One change adds a statement to Bases 3.5.2, ECCS – Operating, to clarify that Condition B of the ITS does not have to be entered if the associated LPI train is unable to support piggyback operation because piggyback operation is not part of the primary success path in CR-3 licensing basis. The Bases will also state that the impact of piggyback operation was already considered in a risk assessment.

Another change proposes to add the word "required" to describe "trains" in the term, "Any combination of three trains inoperable" in current Condition F of ITS 3.6.6. This is proposed to provide consistency with the Bases when referring to any combination of three qualified trains.

No changes to the CR-3 Final Safety Analysis Report (FSAR) are anticipated because of this LAR.

2.0 PROPOSED CHANGES

This LAR proposes changes that either are based on TSTF-430, Revision 2 or TSTF-439-A, Revision 2, or are editorial or administrative in nature. They are presented and discussed separately, below.

Changes Proposed in Accordance with TSTF-430, Revision 2

Consistent with the NRC-approved Revision 2 of TSTF-430 (Reference 8.1), proposed changes to the CR-3 ITS are as follows:

- The addition of new Condition A, one LPI subsystem inoperable with a Completion Time of seven days, to the Actions for ITS 3.5.2, ECCS – Operating. The existing Conditions were relettered accordingly, and the Bases were revised to reflect implementation of this change.
- The addition of new Condition D, one BS train inoperable coincident with one required containment cooling train inoperable with a Completion Time of 72 hours, to the Actions for ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems. The existing Conditions were relettered accordingly, and the Bases were revised to reflect implementation of this change.
- Condition E, newly lettered Condition F, is revised to reflect when the new Condition D is not met also. Note that Required Action F.2 does not need to be relettered. It was incorrectly labeled as F.2 in Amendment 149.
- Condition F, newly lettered Condition G,
- Extension of AOT from 72 hours to seven days for:
    - Condition A, One BS train inoperable in ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems,
    - Condition A, One DC train inoperable in ITS 3.7.8, Decay Heat Closed Cycle Cooling Water System, and
    - Condition A, One RW train inoperable in ITS 3.7.10, Decay Heat Seawater System.
- The revision of the Bases of the affected sections to reflect the changes mentioned above. Note that text in the proposed revision to Bases 3.6.6, Reactor Building Spray and Containment Cooling Systems, Actions D.1 and D.2, differs slightly from the text in TSTF-430, Revision 2 (Reference 8.1) which requires "one of the required containment cooling trains" to be restored to operable. The proposed text requires "one of the required *inoperable* trains" to be restored to operable. This revision of the text was necessary for the Bases to be consistent with the proposed ITS, consistent with TSTF-430, Revision 2.

ITS 3.5.2, Condition A and the AOT of seven days is added to be consistent with the supporting analyses. Specification 3.6.6, Condition D is added to be consistent with the supporting analyses, which did not evaluate the concurrent inoperabilities of one BS train coincident with one containment cooling train. Therefore, the current AOT of 72 hours is retained in the added Condition D and the subsequent Actions are relettered.

Changes Proposed in Accordance with TSTF-439-A, Revision 2

Consistent with the NRC-approved Revision 2 of TSTF-439-A (Reference 8.2), proposed changes to the CR-3 ITS are as follows:

- The removal of the logical connector and the second Completion Times from the following:
  - Conditions A and B of Example 1.3-3, Completion Times,
  - Conditions A and C of ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems,
  - Conditions A and B of ITS 3.7.5, Emergency Feedwater (EFW) System,
  - Conditions A and B of ITS 3.8.1, AC Sources – Operating, and
  - Conditions A, B and C of ITS 3.8.9, Distribution Systems – Operating.
- The removal of text from ITS Section 1.3 discussing second Completion Times.
- The addition of text to ITS Section 1.3 stating that administrative controls are in place that limit the maximum time allowed for any combination of Conditions that could result in a single contiguous occurrence of failing to meet the LCO, and that these controls shall ensure that Completion Times are not inappropriately extended.
- The revision of the Bases of the affected sections to reflect the changes mentioned above.

Editorial/Administrative Changes

This LAR also proposes some changes that are editorial or administrative in nature. These are:

- The removal of Completion Time footnotes that are now obsolete from the following:
  - ITS 3.5.2, ECCS – Operating,
  - ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems,
  - ITS 3.7.5, Emergency Feedwater (EFW) System,
  - ITS 3.7.8, Decay Heat Closed Cycle Cooling Water System, and
  - ITS 3.7.10, Decay Heat Seawater System.
- The revision of the Bases of the affected sections to remove information relevant to the footnotes mentioned above.
- The deletion of Condition 2.C.(13) from the CR-3 Operating License.
- The addition to Bases 3.5.2, ECCS – Operating, of the following text:

  > "Similarly, Condition B does not have to be entered when an associated LPI train is unable to support HPI piggyback operation. The risk associated with this configuration was considered in the development of Condition A."
- The addition of the word "required" to current Condition F of ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems, such that the proposed text will read, "... <u>OR</u> Any combination of three *required* trains inoperable."

3.0 BACKGROUND

Changes Proposed in Accordance with TSTF-430, Revision 2

At CR-3, the LPI serves a dual function as a part of the Decay Heat Removal (DHR) System, and as a part of the ECCS in the emergency operating mode. The DHR System and LPI are combined and share most components including pumps, valves, and piping. The function of the LPI portion of the ECCS is to flood the core with borated water immediately following a large or intermediate loss-of-coolant accident (LOCA) to prevent a significant amount of cladding failure and subsequent release of fission products into the containment. The DHR System is a high-capacity, low-head system with separation and sufficient number of components to provide two-train redundancy for the safeguards mode of operation. It also removes heat from the core for extended periods of time following a LOCA, and in non-emergency conditions such as shutdown and refueling operations.

The BS System removes heat and fission products from the post-accident reactor building atmosphere by directing borated water spray into the reactor building following a LOCA. The system consists of two pumps, two spray headers, and necessary piping, valves, instrumentation, and controls. The LPI and BS System are related in that they both take suction from the borated water storage tank (BWST) and can also draw suction from the reactor building sump for coolant recirculation. When the water in the BWST reaches a low level during the injection mode, the recirculation mode is initiated by realigning the LPI/BS pump suction from the BWST to the reactor building emergency sump. Each LPI train shares common suction piping with its corresponding BS train.

The ITS that is impacted by this probabilistic risk assessment (PRA) is the portion of ITS 3.5.2, ECCS – Operating, that pertains to the AOT for one inoperable train of LPI. Currently, the ITS requires that two trains of ECCS be operable with one train allowed inoperable for 72 hours. It is proposed that the LPI be split out from the ECCS AOT, using the proposed seven day AOT for LPI and retaining the current 72 hours for any ECCS train inoperability other than an inoperable LPI train. The seven day AOT will apply when the LPI train is the only reason for the inoperability of an ECCS train.

At times, one train of BS is impacted by LPI train maintenance because of the common suction piping. Therefore, the ITS change request also includes a proposal to extend the AOT for one inoperable train of BS to seven days.

The DC System facilitates the removal of decay heat from the reactor core. The system also removes process and operating heat from safety related components associated with decay heat removal during normal plant cooldown and following a transient or accident. During plant cooldown below approximately 250°F, the DC System provides core heat removal by transferring heat from the DHR System to the RW System. The DC System has two independent and redundant trains, each capable of supplying 100 percent of the required normal and post-accident cooling. Each train contains a pump, a surge tank pressurized with nitrogen for volume and pressure control, and a heat exchanger which removes heat from the DHR System and rejects it to the RW System.

The CR-3 ITS currently requires two ECCS trains, two BS trains, two containment cooling trains, two DC trains, and two RW trains to be operable. Under the proposed change, an inoperable train of ECCS (which includes the LPI), BS, DC or RW System must be restored to operable status within seven days. In the condition with any one of these trains inoperable, the remaining operable train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining train could result in a loss of function. The seven day AOT is reasonable to perform corrective maintenance on the inoperable train. The proposed seven day AOT is based on the findings of the deterministic and probabilistic analysis in Reference 8.3. This analysis concluded that extending these AOTs to seven days provides plant operational flexibility while simultaneously reducing overall plant risk. This reduction is because the risk incurred by having any one of these trains unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations. Reference 8.3 shows this quantitatively.

Specification 3.6.6, Condition D is added to be consistent with the supporting analyses, which did not evaluate the concurrent inoperabilities of one BS train coincident with one containment cooling train. Therefore, the current AOT of 72 hours is retained in the added Condition D and the subsequent Conditions are relettered.

Risk-informed improvements to technical specifications (TS) are intended to maintain or improve safety while reducing unnecessary burden, and to bring TS into congruence with the Commission's other risk-informed regulatory requirements, in particular the risk assessment and management requirements of 10 CFR 50.65(a)(4).

TS have taken advantage of risk technology as experience and capability have increased. Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on PRA insights. In its final policy statement on TS improvements of July 22, 1993, the Commission stated that it expects that licensees will utilize any plant specific PRA or risk survey in preparing their TS related submittals. The Commission reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encourage greater use of PRA to improve safety decision making and regulatory efficiency. Since that time, the industry and the NRC have been pursuing increased use of PRA in developing improvements to TS.

Changes Proposed in Accordance with TSTF-439-A, Revision 2

Changes proposed under TSTF-439-A, Revision 2 eliminates the second Completion Times for five of the CR-3 ITS:

- ITS 1.3, Example 1.3-3, Completion Times,
- ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems,
- ITS 3.7.5, Emergency Feedwater (EFW) System,
- ITS 3.8.1, AC Sources – Operating, and
- ITS 3.8.9, Distribution Systems – Operating.

The concern is that these ITS could theoretically allow indefinite operation of the plant while failing to meet an LCO. If an LCO requires operability of two systems, it is possible to enter the Condition for one inoperable system and before restoring the first system, the second system becomes inoperable. With the second system inoperable, the first system is restored to operable status. Before restoring the second system, the first can become inoperable again, and so on. Under this scenario, it is theoretically possible to operate indefinitely without ever satisfying the LCO. This also could occur with LCOs which require only one system to be operable, and the Conditions describe two or more mutually exclusive causes of inoperability.

An NRC internal memo discussing the issue, dated August 5, 1991, stated:

> "In these Specifications the following phrase was added in the Completion Time column of the Conditions that could extend the AOT: '[10 days] from discovery of failure to meet the LCO.' The [10 day] Completion Time cap is found by adding the maximum Completion Times from the two Conditions that could extend the AOT."

The decision to add this second Completion Time was later documented in a memo from the NRC to the industry lead plant representatives dated December 16, 1991.

This issue of "flip flopping" between Conditions only applies if the LCO is not met. If the LCO requirements are met, even for an instant, this issue does not occur. This is a highly unlikely scenario and the Industry argued that it would never occur, but the NRC believed it should be addressed when developing the Improved Standard Technical Specifications (ISTS) because no other regulatory process was in place at that time to prevent or respond to such a situation, should it occur.

The addition of second Completion Times did not originally create an operational restriction because the likelihood of experiencing concurrent failures such that the second Completion Time is limiting is very remote. However, these second Completion Times became a problem when the Industry proposed risk-informed Completion Times for some specifications which contained second Completion Times. Specifically TSTF-409, Containment Spray System Completion Time Extension, and TSTF-430, AOT Extension to 7 Days for LPI and Containment Spray (Reference 8.1) proposed the extension of a Completion Time. Following the methodology described in the August 5, 1991 memo, the second Completion Time was extended by the same amount (i.e., the second Completion Time was set equal to the sum of the two Completion Times). However, in letters to the TSTF dated November 15, 2001 and September 10, 2002, the NRC stated that the extension of the second Completion Times in TSTF-409 and TSTF-430 was inappropriate because of the two Completion Times added to obtain the second Completion Time limit, one was risk based and the other was deterministic. Eventually, the NRC accepted the addition of these two Completion Times and TSTF-409 and TSTF-430 were approved. However, second Completion Times complicate the presentation of the CR-3 ITS and the implementation of risk-informed Completion Times. In addition, other regulatory requirements [most notably 10 CFR 50.65, Maintenance Rule], not present when the US NRC Regulations on ISTS were originally developed, eliminate the need for these second Completion Times.

Editorial/Administrative Changes

There are essentially three changes proposed in this LAR which are considered editorial or administrative in nature. One involves five specifications, their Bases and the Operating License. This proposed change is the removal of obsolete footnotes and language referring to these footnotes in the Bases and the Operating License. The footnotes were implemented in Amendments 212, 215, and 221 (dated May 18, 2004, January 11, 2005, and September 15, 2005, respectively) to extend the AOT for certain equipment on a one-time only basis to accommodate some maintenance. Since this work has been completed, the footnotes are no longer meaningful and can be deleted.

The second proposed change adds the word "required" to clarify which trains are the subject of a Condition statement. ITS 3.6.6 describes the requirements of the BS and Containment Cooling Systems. Condition F (which will be proposed Condition G due to an unrelated proposed change) concerns the condition for two BS trains inoperable OR any combination of three trains inoperable. The Bases state that the second part of this Condition applies to any combination of three or more Reactor Building Spray and *required* Containment Cooling trains being inoperable. To eliminate any ambiguity between the Bases and the Specification, the addition of the word "required" to the second part of this Condition is proposed such that it now reads:

Two reactor building
spray trains
inoperable.

OR

Any combination of
three required trains
inoperable.

The third change adds a clarifying statement to Bases 3.5.2, ECCS – Operating. The existing Bases discuss why it is unnecessary to enter LCO 3.5.2 when a train of the recirculation line to the RB sump is unavailable. The new text provides similar clarification regarding operability when an LPI train is unable to support HPI piggyback operation:

> "Similarly, Condition B does not have to be entered when an associated LPI train is unable to support HPI piggyback operation. The risk associated with this configuration was considered in the development of Condition A."

4.0 TECHNICAL ANALYSIS

Changes Proposed in Accordance with TSTF-430, Revision 2

An analysis for this change prepared by Framatome for the B&W Owners Group is documented in the Reference 8.3 topical report. This was approved by the NRC on July

15, 1999. The NRC's evaluation of this report considered both deterministic and probabilistic analyses. These are discussed below.

<u>Deterministic: Effect on Safety Analyses</u>

The deterministic evaluation presented in the Topical Report (Reference 8.3) for CR-3 consisted of a review of plant systems and safety functions affected by the entry into each AOT. FPC identified and quantitatively and qualitatively assessed the safety functions associated with the systems affected by the AOTs. FPC determined there are no structures, systems, or components (SSCs) that will change status such that they become significant to public health and safety due to the proposed change. No new accidents or transients are introduced by the proposed change. No physical changes are being made to any of the systems affected by these AOTs. The function and operation of these systems will remain the same as that described in the FSAR. Protective measures will be taken to ensure that unanticipated compromises to system redundancy, independence, and diversity will not occur during maintenance activities. These protective measures will continue after the proposed AOT has been implemented. The impact of the proposed changes on the safety margins was also considered. Extending the AOT to seven days for one inoperable train does not affect any assumptions or inputs in the FSAR. The NRC found the deterministic evaluation acceptable.

<u>Probabilistic: Effect on Risk Informed Analysis</u>

The PRA used in the Reference 8.3 topical report to assess the impact of the proposed change is based upon similar measures defined in Regulatory Guides (RGs) 1.174 and RG 1.177 (References 8.8 and 8.9, respectively).

The risk impacts of the proposed change were calculated in Reference 8.3 and compared against the acceptability guidelines as stated in the RGs. The CR-3 base core damage frequency (CDF) from internal events calculated in the topical report is 6.35E-06/yr. The overall CDF for the proposed change was 7.28E-06/yr for maintenance using the full AOT. This resulted in an incremental CDF for the proposed change of 9.3E-07/yr. In sensitivity analyses, the incremental or increase in CDF did not change appreciably when external events were included. More results are presented in Reference 8.3.

The CR-3 base large early release frequency (LERF) from internal events is 1.84E-07/yr. The overall LERF for the proposed change was 1.93E-07/yr for maintenance using the expected mean duration, and 2.29E-07/yr using the full seven day duration. This resulted in a ΔLERF for the proposed change of 9.0E-09/yr and 4.5E-08/yr, respectively. In sensitivity analyses, the incremental or increase in LERF did not change appreciably when external events were included. More results are presented in Reference 8.3.

The calculated value of incremental conditional core damage probability (ICCDP) for the proposed change was 6.6E-07. The calculated value of incremental conditional large early release probability (ICLERP) for the proposed change was 1.5E-08. These values were found to be acceptable by the NRC assuming the following compensatory measures were taken to lower the risk impacts:

- Avoiding simultaneous outages of additional risk-significant components during the AOT of the LPI and BS System trains (and in the case of CR-3, the RW and DC System trains, as well). The components whose simultaneous outages are to be avoided, in addition to the current TS requirements, include both trains of the EFW, both High Pressure Injection (HPI) trains (for reasons other than being inoperable due to the associated LPI train), Reactor Building Cooling Unit, and their power supplies.
- Defining specific criteria for scheduling only those preventive maintenance procedures which can be completed within the AOT, such that the chance for a forced outage due to failure to complete the maintenance is negligible.
- Assuring that the frequency of entry into the AOT, and consequently, the average maintenance duration per year, remain within that assumed in this submittal.
- Taking measures to ensure that while maintaining the LPI or BS Systems (and in the case of CR-3, the RW and DC Systems, as well), both trains are not made unavailable unless it is necessary. In some situations, maintenance in one of the trains can be conducted without affecting the other train.

These four compensatory measures will be put in place prior to implementing the revision to the CR-3 ITS.

The NRC's Safety Evaluation Report also required the implementation of a "Configuration Risk Management Program" which meets criteria put forth in the Safety Evaluation. Maintenance Rule, 10 CFR 50.65(a)(4), eliminates the need for a "Configuration Risk Management Program" to support this change since 10 CFR 50.65(a)(4) mandates a similar program.

The Reference 8.3 topical report describes the risk informed evaluation that was performed to evaluate the extension of the AOT for LPI, BS, DC, and RW System to seven days. The NRC found this evaluation acceptable, as detailed in their Safety Evaluation (Reference 8.4). In this review, the NRC requested that any submittals based on References 8.1 and 8.3 include information on PRA quality to ensure that specific PRAs are adequate to support the proposed changes. Specifically, the following information was requested:

- Verification that the PRA reflects the as-built, as-operated plant.
- Updates of the PRA since the last review cycle, including corrections of weaknesses identified by past reviews.
- Details of their peer review process, a summary of the peer review findings, and a discussion of this independence of internal reviews/reviewers.
- Description of PRA quality assurance methods.
- Results of reviews of pertinent accident sequences and cut sets for irregularities (with respect to this application).

An analysis was performed by FPC consistent with NRC RGs 1.174 and RG 1.177 (References 8.8 and 8.9, respectively) to calculate the quantitative impact of a proposed permanent risk informed ITS change for the LPI, BS, DC and RW Systems from 72 hours to seven days. The analysis is included at Attachment E to this submittal. This analysis is plant-specific using the CR-3 Probabilistic Safety Assessment (PSA) model for on-going operation.

The following information is provided to address the issues requested in the NRC's Safety Evaluation.

Quality of the CR-3 PSA

The base case Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) values for CR-3 have been calculated by FPC to be:

- CDF = 4.99E-06/yr
- LERF = 3.69E-07/yr

These values are generally lower than the industry average, in part, for the following reasons:

- Byron Jackson N-9000 Reactor Coolant Pump (RCP) seals are installed and are assumed to maintain their integrity as long as they have seal injection or seal cooling, or the RCPs are tripped. This greatly reduces the likelihood of an RCP seal failure causing a LOCA.
- Offsite power is supplied from a 230 kilovolt (kV) switchyard that has feeds from the grid and from three fossil plants onsite. CR-3 outputs to a separate 500 kV switchyard. Based on this, dependent loss of offsite power (LOOP) events occurring due to trip initiators is not considered a credible event.
- CR-3 has a third non-safety related diesel that can power an Engineered Safeguards (ES) bus that adds additional redundancy for LOOP scenarios.
- CR-3 maintains a diverse secondary cooling capability, including automatically actuated steam and diesel driven emergency feedwater pumps, a backup motor driven pump powered from the ES bus, and a backup motor driven pump that is powered from normal offsite power or the alternate AC diesel generator.
- CR-3 has three high head injection/makeup pumps each capable of providing adequate primary cooling via the pressurizer power-operated relief valve or pressurizer safeties at full Reactor Coolant System (RCS) pressure. The High Pressure Injection (HPI) pumps also have diverse support systems. Two of the pumps have backup cooling and one can be powered from either ES 4160 kV bus.
- CR-3 has separate safety-related service water (RW) systems for the decay heat removal system and nuclear services support for other systems. The nuclear services system also has a third non-safety related train that can cool normal loads.
- CR-3 has a dedicated chiller installed for 10CFR50 Appendix R (fire) considerations that is not dependent on SW.

The PSA inputs used for this application were generated using updated Individual Plant Examination (IPE) models developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and associated supplements. The original development work was a Level 1 PRA study completed in 1987 (Reference 8.10), which was submitted to the NRC and reviewed by Argonne National Laboratory (NUREG/CR-5245). This study was subsequently updated for the Generic Letter 88-20 IPE submittal to include a Level 2 containment analysis and an internal flooding analysis. The study was subject to reviews by the relevant CR-3 system engineers, and review of the event sequence analysis, quantification, and recovery analysis by the Nuclear Safety

Supervisor at CR-3, a former Senior Reactor Operator.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. These changes have been made by individuals knowledgeable in risk assessment techniques and methods, and reviewed by plant Engineering and Operations personnel familiar with the plant design and operation. The current PSA model and the risk assessment performed for this application have been documented as a calculation.

Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by a PSA engineer, reviewed by another PSA engineer, and approved by the PSA Supervisor or designee. Procedures, PSA model documentation and associated records for applications of the PSA models are controlled documents.

Since the submittal of the original PRA study in 1987, the PSA models have been maintained consistent with the current plant configuration such that they are considered "living" models which reasonably reflect the as-built, as-operated plant. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PSA technology. The update process ensures that the applicable changes are implemented and documented in a timely manner so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance and update process is described in administrative procedure ADM-NGGC-0004, "Updates to PSA Models." Guidance to determine the need for a model update is provided in this procedure. Each PRA model update is documented in accordance with plant procedures governing the preparation of engineering calculations, which includes an independent review of each calculation that is an input to the PRA model of record.

PSA Software

Computer programs that process PSA model inputs are verified and validated in accordance with administrative procedure CSP-NGGC-2505, "Software Quality Assurance and Configuration Control of Business Computer Systems." This procedure provides for software verification and validation to ensure the software meets the software requirement specifications and functional requirements, and typically includes a comparison of results with those generated from previously approved software.

Model Changes Since Submittal of the IPE

Since the submittal of the IPE, there have been several significant plant design changes incorporated into the PSA model that have resulted in a reduction in the CDF. A summary of significant model changes incorporated due to these plant changes includes the following:

- Backup ES Transformer added (A and B safeguards trains powered from separate transformers)

- FWP-7 powered by offsite power or the alternate AC diesel generator
- Appendix R chiller installed
- EFP-3 installed
- Installed Alternate AC diesel, which can power either Essential Bus or FWP-7
- Low pressure injection BWST suction valves changed to normally open
- HPI discharge throttle valves and cross-ties added
- Revision of emergency operating procedures reflected in human action probabilities

In addition to these plant changes, updates have been made to plant-specific data (through 1999) and initiating events data, as well as updates to the methods used for human reliability, common cause, internal flooding, and Level 2 analyses.

As of the date of this submittal, there are no outstanding plant changes which would require a change to the PSA model, and no planned plant changes which would be implemented prior to the fall 2007 refueling outage which would require a change to the PSA model.

PSA Reviews

As discussed above, the original CR-3 PRA study was reviewed by Argonne National Laboratory as documented in NUREG/CR-5245. For the IPE submittal, multiple levels of review were used, including an assessment by Engineering and Operations personnel familiar with the plant design and operation. Subsequent revisions to the PSA models were performed by qualified individuals with knowledge of PSA methods and plant systems. Involvement by Engineering and Operations personnel in providing input and review of results was obtained when required based on the scope of the changes being implemented.

The CR-3 PSA model and documentation was subjected to the industry peer certification review process in September 2001. The industry peer certification review was conducted by a diverse group of PSA engineers from other Babcock & Wilcox (B&W) plants, industry PSA consultants familiar with the B&W plant design, and a representative from the Institute of Nuclear Power Operations (INPO). The reviewers involved in the peer review process were not employees of the company, and were not involved with the development of the PRA model. The certification review covered all aspects of the PSA model and the administrative processes used to maintain and update the model. This review generated specific recommendations for model changes to correct errors, as well as guidance for improvements to processes and methodologies used in the CR-3 PSA model, and enhancements to the documentation of the model and the administrative procedures used for model updates.

Following completion of this review, the CR-3 PSA model was revised to address each issue identified which affected the model. The significant changes identified included:

- Update of the plant-specific thermal-hydraulic analyses that provide the bases for accident sequences, system success criteria, and timing for operator actions.
- Revision of accident sequence logic for steam generator tube rupture (SGTR) and anticipated transient without scram (ATWS) mitigation.

- Development of an initiating event to address the loss of all raw water pumps (loss of ultimate heat sink).
- Update of the interfacing systems loss of coolant accident (ISLOCA) analyses.
- Update of the human reliability analysis including the dependency analysis for multiple operator action responses to an event, and
- Update of the Level 2 analysis.

All peer review items which affect the PSA model have been addressed and are reflected in the PSA model used in this submittal.

At the time of the peer review, the Level 2 model was not yet completed, and only a preliminary draft version, along with the original IPE Level 2 results, were available for review. The Level 2 model is now complete, and the findings identified from the peer certification review of the preliminary results and the IPE model have been addressed.

General Assumptions

Some general assumptions made in the FPC analysis are:

- The maximum allowed AOT times for the LPI, BS, DC and RW systems will be seven days.
- The extended AOT used for each train will be once per operating year on average.
- The extended AOT will not be performed on trains A and B simultaneously.
- The extended AOT occurs when the plant is in operating Mode 1.
- It is assumed that the AOT maintenance occurs simultaneously on the LPI, BS, DC and RW systems.
- During performance of the AOT, the corresponding opposite train equipment and diesel are considered to be protected.
- Shutdown risk was not considered in the delta risk evaluation to ensure results are bounding. Because a direct comparison of online to offline risk is difficult to quantify, the shutdown risk of having a LPI train unavailable is assumed to be zero.
- All other limiting conditions will remain unchanged.
- External events can be evaluated qualitatively.

Results

The results of the cutsets for the CDF, LERF, and various sensitivity cases were each reviewed in order to assure that the resulting accident sequences gave reasonable results for the various configurations examined for this study. The following conclusions are made:

- The delta CDF of 5.5E-07/yr is below the RG 1.174 threshold of 1E-06/yr and is considered low risk.
- The delta LERF of 1E-09/yr is below the RG 1.174 threshold of 1E-07/yr and is considered low risk.
- The ICCDP values for trains A and B are calculated to be 2.97E-07 and 4.05E-07, respectively. These values are below the RG 1.177 threshold for ICCDP of 5.0E-07 and are considered small.

- The Incremental Conditional Large Early Release Probability (ICLERP) values for trains A and B are calculated to be 2.88E-10 and 4.03E-10, respectively. These values are below the RG 1.177 threshold for ICLERP of 5E-08 and are considered small.

The results of the analysis show that the risk during shutdown is reduced by doing maintenance on the Decay Heat Removal System while at power rather than during shutdown when it is the primary source of cooling. Also, as discussed in the topical report (Reference 8.3) preventative maintenance can be scheduled more frequently, enhancing the overall reliability of equipment and reducing the number of entries into LCOs. A longer AOT also allows more flexibility in work scheduling which leads to more orderly completion of maintenance.

The CR-3 IPE and supporting data were reviewed to identify external events that could influence the risk for the subject activities. The only potentially significant external events are fires and severe weather. Each is discussed below.

Fire Risk Sensitivity

A fire risk sensitivity study and a qualitative assessment for comparing shutdown verses online risk have been performed to provide additional insight. CR-3 does not have a fire PRA model that can be used to qualify the effect of the postulated fire scenario on LERF. However, because the predominant contributors to LERF for CR-3 are scenarios based on SGTRs or ISLOCAs, the LERF impact is estimated to be very low because the LPI, BS, DC and RW systems are not significant mitigating systems for SGTR sequences. Therefore, any increase in LERF due to fire will be very small. Even a fire that affects both trains or manual actions credited in the Fire Study, the delta risk is expected to be minimal. The greatest risk impact due to Train A being out of service is expected for fires which affect Train B equipment. Similarly, the greatest risk impact when Train B is out of service is a fire that affects Train A equipment.

The instantaneous CDF due to fire with Train A out of service is 1.39E-04/yr and with Train B out of service, 2.49E-04/yr. This translates into an estimated Conditional Core Damage Probability (CCDP) due to fire with the seven day AOT of 2.67E-06 for Train A and 4.78E-06 for Train B. These values are considered to be bounding because it was conservatively assumed there was a 0.1 chance of core damage occurring given that a fire occurred. Additional compensatory actions such as dedicated fire watches would further reduce this value.

A comparison of the online and offline risk can not be directly quantified, but can be addressed qualitatively. Although there is some level of increased risk performing this work online, the shutdown fire risk is expected to be greater due to transient initiating events. The risk due to fire is dominated by these initiating events. During outages, transient initiating event frequencies increase by an order of magnitude due to increased storage of flammable material in the station and increased maintenance. Furthermore, fire suppression by existing installed equipment or a fire brigade is impaired by maintenance activities that limit accessibility to the fire areas by staged equipment and scaffolding.

Weather Sensitivity

The main impact of a severe weather event to CR-3 is an increased probability for a LOOP. FPC performed a sensitivity case that demonstrated a minimal increase in risk due to a higher LOOP frequency during the extended AOT. To evaluate the sensitivity to weather events, the frequency of losing offsite power was increased in the PRA model by a factor of three. This gave a new base CDF of 5.46E-06/year. With Train A in the proposed AOT, the CDF is 2.20E-05/year. With Train B in the proposed AOT, the CDF is 2.85E-05/year. These values translate to an ICCDP for Train A of 3.16E-07, and for Train B of 4.41E-07, and a delta ICCDP for Train A of 1.90E-08, and for Train B of 3.60E-08.

The delta ICCDP due to the increase in the LOOP initiating events was less than 1E-07 which shows the results are not sensitive to external events that increase the frequency of a LOOP.

Raw Water Vault Maintenance Sensitivity

A sensitivity analysis was done by FPC to determine if RW vault work should be avoided during the seven day AOT. RW vault work would remove additional pumps from service (i.e., RWP-2A / RWP-3A for work done in train A RW vault and RWP-2B / RWP-3B / RWP-1 for work done in Train B RW vault). This analysis compares the ICCDP of the proposed seven day AOT and a three day RW vault outage that does not occur during that seven day period with the ICCDP of the proposed seven day AOT overlapping a three day RW vault outage. The ICCDP for Train A with the AOT scheduled separate from the RW vault outage is 4.69E-07, and with the AOT scheduled concurrent with the RW vault outage is 3.42E-07. The ICCDP for Train B with the AOT scheduled separate from the RW vault outage is 8.29E-07, and with the AOT scheduled concurrent with the RW vault outage is 6.59E-07. This shows that if maintenance is required on the RW vaults and a decay heat outage is also being planned, performing these activities concurrently would minimize the risk to the public.

Compensatory Measures

In Reference 8.4, the NRC stated that the compensatory measures defined as part of the proposed AOT changes were applicable and would reduce the risk impact during the AOT to more acceptable levels. For the performance of maintenance on LPI, BS, DC or RW Systems planned to extend beyond the current 72 hours allowed in the ITS, CR-3 will take these additional precautions to minimize risk:

- CR-3 will perform procedure CP-253, "Power Operation Risk Assessment and Management," which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on CDF. CR-3 will avoid an AOT for the LPI, BS, DC, or RW Systems that results in "Higher Risk" (Orange Color Code). These protective measures currently ensure that unanticipated compromises to system redundancy, independence, and diversity will not occur during maintenance activities. These protective measures

- Simultaneous outages of additional risk-significant components will be avoided during the AOT of LPI, BS, DC or RW trains. These risk-significant components whose simultaneous outages are identified to be avoided, in addition to current ITS requirements, are EFW, Auxiliary Feedwater System, Emergency Feedwater Initiation and Control System, HPI, Appendix R Cooler, and their power supplies.
- Specific criteria are already defined for scheduling only those preventive maintenances which can be completed within the AOT, such that the chance for needing a forced outage for failing to complete the maintenance is negligible. CR-3 currently uses procedures WCP-100, "On-Line Planning, Scheduling, and Implementation," ADM-NGGC-0102, "Project Review and Authorization and Long Range Planning," and ADM-NGGC-0104, "Work Management Process," to schedule work to increase equipment reliability and optimize system availability by taking advantage of online system outage opportunities.
- CR-3 will not initiate an extended preventive maintenance outage on the LPI, BS, DC, or RW Systems if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.
- A periodic fire watch will be established and transient combustibles will be limited in the decay heat pump vault (which houses the decay heat and BS pumps) of the opposite train, and in the seawater room (which houses the RW and SW pumps) of the opposite train.

In addition to the compensatory measures above, the following actions will also be taken:

- The frequency of entry into the Completion Time, and consequently the average maintenance duration per year, will be watched to assure they remain within that assumed in the submittal. The CR-3 Maintenance Rule Committee, who is responsible for monitoring this, has a PRA Engineer as a member. The role of the PRA Engineer, in part, is to verify the accuracy of the assumptions in the model. This includes verifying that the average maintenance duration per year remains within that assumed in this submittal.
- If maintenance is required on the RW vaults and a decay heat outage is also being planned, these activities should be scheduled concurrently, if possible.

Changes Proposed in Accordance with TSTF-439-A, Revision 2

The adoption of a second Completion Time was based on an NRC concern that in a certain scenario, a plant could continue to operate indefinitely without complying with a safety significant LCO. If separate ITS Conditions are entered and exited such that they overlap, the LCO may never be met even though no Completion Time is violated. In 1991, the NRC could not identify any regulatory requirement or program which could prevent this misuse of the ITS. However, this is no longer the case. Two programs provide a strong disincentive to continue operation with multiple concurrent inoperabilities of the type the second Completion Times were designed to prevent.

The Maintenance Rule: 10 CFR 50.65 (a)(1), the Maintenance Rule, requires CR-3 to monitor the performance or condition of SSCs against licensee-established goals to ensure that the SSCs are capable of fulfilling their intended functions. If the performance or condition of an SSC does not meet established goals, appropriate corrective action is required to be taken. The NRC Resident Inspectors monitor the CR-3 Corrective Action process and can take action if the maintenance program allowed the systems required by a single LCO to become concurrently inoperable multiple times. The performance and condition monitoring activities required by 10 CFR 50.65 (a)(1) and (a)(2) will identify if poor maintenance practices resulted in multiple entries into the Actions of the CR-3 ITS and created unacceptable unavailability of these SSCs. The effectiveness of these performance monitoring activities, and associated corrective actions is evaluated at least every refueling cycle, not to exceed 24 months per 10 CFR 50.65 (a)(3).

Under the CR-3 ITS, the Completion Time for one system is not affected by other inoperable equipment. The second Completion Times were an attempt to influence the Completion Time for one system based on the condition of another, if the two systems were required by the same LCO. However, 10 CFR 50.65(a)(4) is a better mechanism to apply this influence as the Maintenance Rule considers all inoperable risk-significant equipment, not just the one or two systems governed by the same LCO.

Under 10 CFR 50.65 (a)(4), the risk impact of all inoperable risk-significant equipment is assessed and managed when performing preventative or corrective maintenance. The risk assessments are conducted using the procedures and guidance endorsed by RG 1.182, "Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants." RG 1.182 endorses the guidance in Section 11 of Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed maintenance is acceptable. This comprehensive program provides much greater assurance of safe plant operation than the second Completion Times in the TS.

The Reactor Oversight Process: Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," describes the tracking and reporting of performance indicators to support the NRC's Reactor Oversight Process (ROP). The NEI document is endorsed by Regulatory Issue Summary (RIS) 2001-11, "Voluntary Submission of Performance Indicator Data." NEI 99-02, Section 2.2, describes the Mitigating Systems Cornerstone. NEI 99-02 specifically addresses Emergency AC Sources (which encompasses the AC Sources and Distribution System LCOs), and the Auxiliary Feedwater System. Extended unavailability of these systems due to multiple entries into the Actions would affect the NRC's evaluation of the licensee's performance under the ROP.

In addition to these programs, the addition of a requirement to Section 1.3 of the CR-3 ITS is proposed that will require CR-3 to have administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls will consider plant risk and limit the maximum contiguous time of failing to meet the LCO. This CR-3 ITS requirement, when considered with the regulatory processes discussed above, provide an equivalent or superior level of plant safety without the unnecessary complication of the CR-3 ITS by second Completion Times.

Each affected ITS is discussed below.

CR-3 ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems

CR-3 ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems, currently has a 72 hour Completion Time for one BS train inoperable (Condition A) and a seven day Completion Time for one required Containment Cooling train inoperable (Condition C). Conditions A and C have a second Completion Time of ten days from discovery of failure to meet the LCO. Below, Figure 1 shows how these Completion Times relate to each other. In the figure, containment spray train refers to BS train.

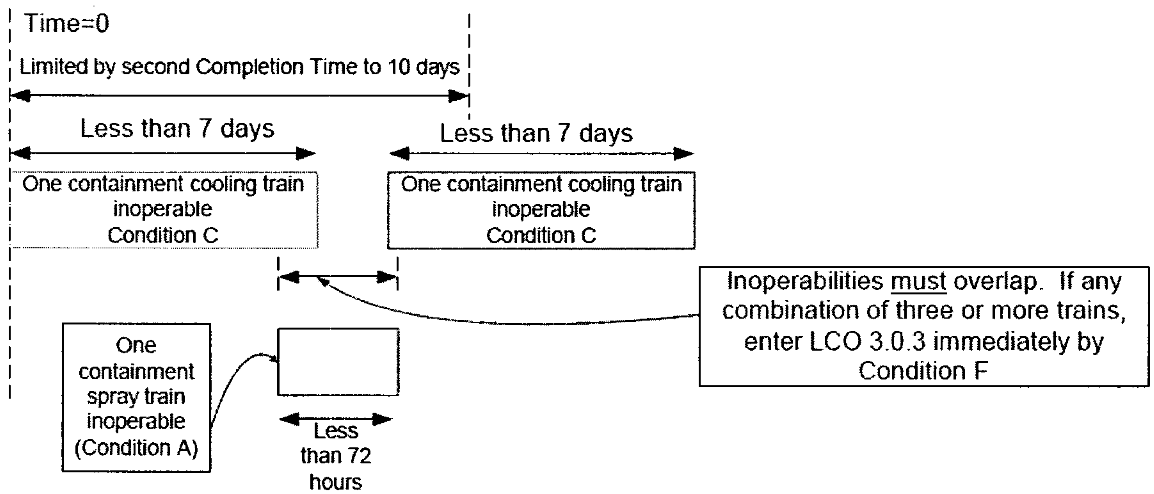

Figure 1

The second Completion Time (the ten days) is not needed. Condition F of CR-3 ITS 3.6.6 states that if two BS trains are inoperable or any combination of three or more trains is inoperable, LCO 3.0.3 must be entered immediately. Therefore, any combination of two of the four trains can perform the safety function. The second Completion Time restricts operation with only one train inoperable, but that is unnecessary because when one train is inoperable, there are still three operable trains and only two are needed to perform the safety function. Therefore, the second Completion Time is overly restrictive.

CR-3 ITS 3.7.5, Emergency Feedwater System

CR-3 ITS 3.7.5, EFW, has a seven day Completion Time for one inoperable steam supply to a turbine driven EFW pump (Condition A) and a 72 hour Completion Time for one EFW train inoperable for reasons other than Condition A (Condition B). Conditions A and B have a second Completion Time of ten days from discovery of failure to meet the LCO. In order for the second Completion Time to be limiting, entry into and out of Conditions A and B must occur, which requires the turbine driven and motor driven EFW pumps to be concurrently inoperable. Below, Figure 2 shows how these Completion Times relate to each other. In the figure, AFW refers to the CR-3 EFW System.

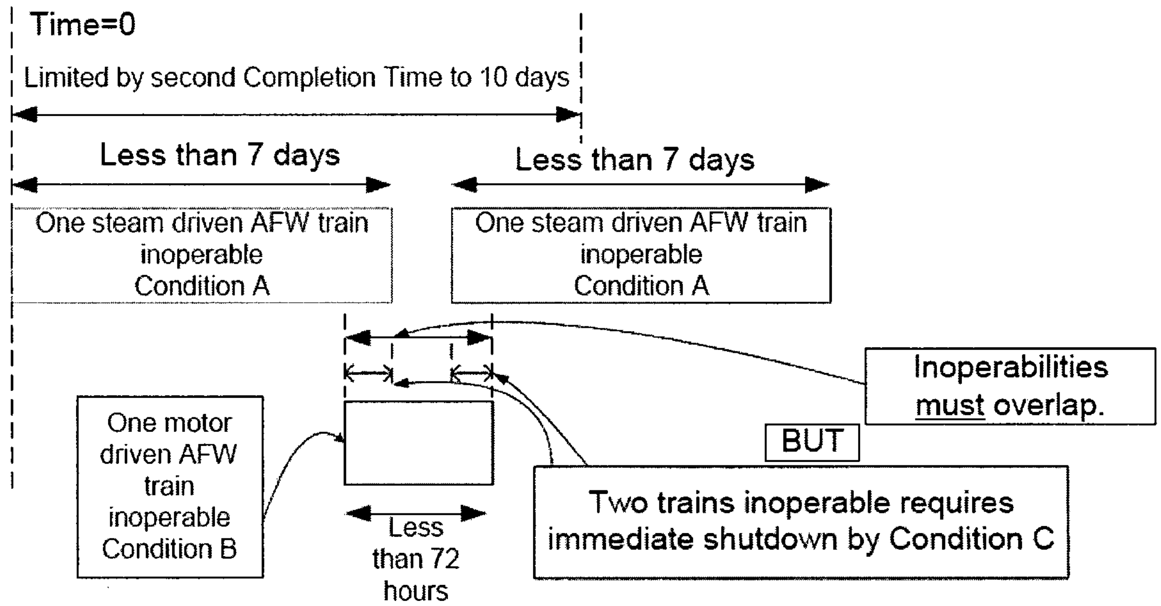

Figure 2

The second Completion Time is not needed. Condition C states that if either EFW train is inoperable, the plant must be in Mode 3 in 6 hours and Mode 4 in 12 hours. For the second Completion Time to be limiting, Conditions A and B must be entered concurrently. However, Condition C requires the plant to initiate action immediately to restore one of the EFW trains to operable. Having failed this, necessary for the second Completion Time to be in effect, the plant must enter LCO 3.0.3. The second Completion Time will never be more limiting than LCO 3.0.3 and therefore can be removed. In addition, the ROP monitors the availability of the EFW System. Such frequent, repeated failures of the EFW System would be reported to the NRC, and this represents a strong disincentive to such operation.

CR-3 ITS 3.8.1, AC Sources - Operating

CR-3 ITS 3.8.1, AC Sources - Operating, has a 72 hour Completion Time for one required offsite circuit inoperable (Condition A), and a 72 hour Completion Time for one Emergency Diesel Generator (EDG) inoperable or 14 days for one EDG inoperable and alternate AC power is available (Condition B). Condition A has a second Completion Time of "6 days from discovery of failure to meet the LCO." Condition B has two second Completion Times: "6 days from discovery of failure to meet LCO" for one EDG inoperable, and "17 days from discovery of failure to meet LCO" for one EDG inoperable and alternate AC Power is available.

For the second Completion Time to be in effect, Condition A or B must be entered. For the second Completion Time to stay in effect, the other Condition must be entered before the first inoperable system is restored. Only then can the first inoperable system be restored. Again, for the second Completion Time to remain in effect, the first system must be declared inoperable before the second system is restored. This highly improbable scenario is further limited by Condition D which applies when both a required offsite circuit and an EDG are inoperable. It limits plant operation in this Condition to 12 hours. Below, Figure 3 shows how these Completion Times relate to each other.

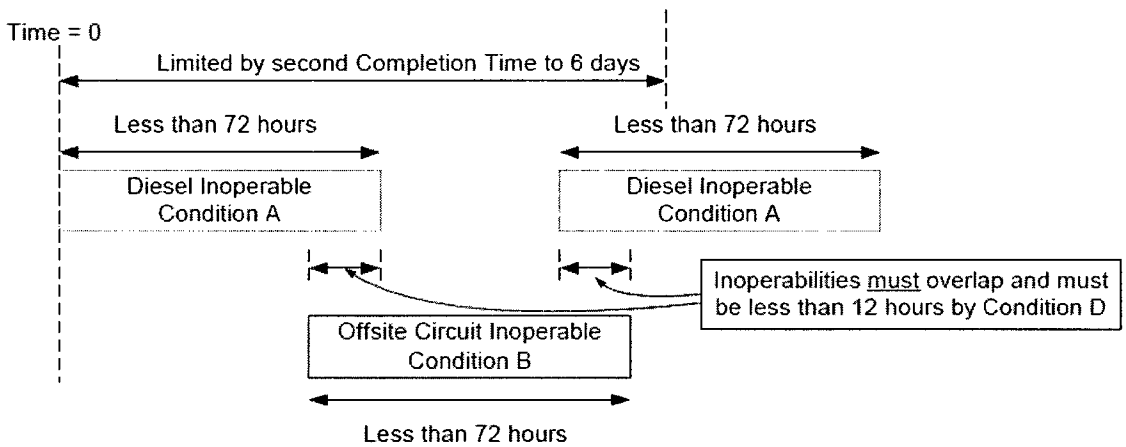

Figure 3

As previously stated, the ROP monitors the availability of mitigating systems, including Emergency AC sources (EDG unavailability). Such frequent, repeated failures of the AC sources would be reported to the NRC, and this represents a strong disincentive to such operation.

<u>CR-3 ITS 3.8.9, Distribution Systems – Operating</u>

CR-3 ITS 3.8.9, Distribution Systems – Operating, has an eight hour Completion Time for one AC electrical power distribution subsystem inoperable (Condition A) or one AC vital bus subsystem inoperable (Condition B), and a two hour Completion Time for one DC electrical power distribution subsystem inoperable (Condition C). Conditions A, B and C each have a second Completion Time of "16 hours from discovery of failure to meet the LCO." Condition E applies if two trains have inoperable distribution subsystems that result in a loss of function. In this condition, then LCO 3.0.3 must be entered immediately. Below, Figure 4 shows how these Completion Times relate to each other.

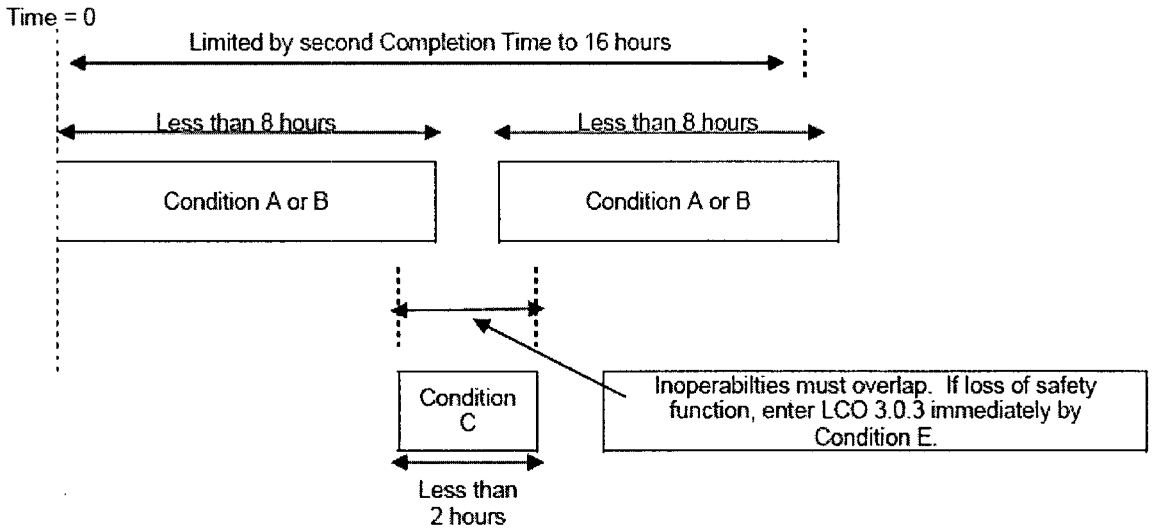

Figure 4

The second Completion Time is not even needed. These Completion Times are very short, two or eight hours. This provides little opportunity to restore systems such that Conditions overlap and multiple inoperabilities occur. Further, any overlapping inoperabilities that occur and result in a loss of safety function require a plant shutdown in accordance with LCO 3.0.3.

Editorial/Administrative Changes

In several CR-3 ITS, Completion Times were footnoted to reflect a one time extension of the AOT to allow for some work done on various components that has since been completed. Since all these footnotes are now obsolete, their deletion has little impact. The specific deletions are as follows:

- CR-3 ITS 3.5.2, ECCS – Operating. The footnote proposed for deletion states that on a one-time basis, an Emergency Core Cooling System train may be inoperable for up to ten days to allow performance of RW System Pump repairs to be done online, and that this footnote expires once the work is complete.
- CR-3 ITS 3.6.6, Reactor Building Spray and Containment Cooling Systems. The footnote proposed for deletion states that a BS train may be inoperable up to ten days to allow performance of RW System pump repairs to be done online, and that this footnote expires once this work is complete.
- CR-3 ITS 3.7.5, Emergency Feedwater (EFW) System. The footnote proposed for deletion states that on a one-time basis, an EFW train may be inoperable for up to 14 days to allow performance of EFW pump repairs. This allowed extension expired on March 31, 2005.
- CR-3 ITS 3.7.8, Decay Heat Closed Cycle Cooling Water (DC) System. The footnote proposed for deletion states that on a one-time basis, a DC train may be inoperable for up to ten days to allow performance of RW System pump repairs to be done online, and that this footnote expires once the work is complete.
- CR-3 ITS 3.7.10, Decay Heat Seawater System. The footnote proposed for deletion states that on a one-time basis, a RW train may be inoperable for up to

ten days to allow performance of RW System pump repairs to be done online, and that this footnote expires once the work is complete.

- The Bases for each of these specifications above contain the same language as in the ITS footnotes. It is proposed to remove this language from each of these Bases, as well.
- When the footnote was added to CR-3 ITS 3.7.9 (CR-3 Amendment Number 212, dated May 18, 2004), the Operating License was revised to reflect the one time extended AOT. Additional Condition 2.C.(13) was added to the CR-3 Operating License to reflect the change and identify compensatory measures. Like the footnote, this information in the Operating License is obsolete. Deletion of Additional Condition 2.C.(13) from the Operating License should also have little impact. The deletion should include sub items (a) through (i) and seven identified fire zones.

Since these statements are obsolete and no longer meaningful, there is no technical justification to retain them in the CR-3 ITS or Operating License.

Another proposed change adds the word "required" to clarify which trains are the subject of a Condition statement. This does not change the Condition statement technically since it is just making it consistent with the Babcock and Wilcox Standard Technical Specifications, NUREG-1430.

The final editorial/administrative change adds text to Bases 3.5.2, ECCS – Operating, to clarify that Condition B of the ITS does not have to be entered when an associated LPI is unable to support piggyback operation. Because it was already considered in the risk assessment supporting the proposed ITS change.

5.0 <u>REGULATORY ANALYSIS</u>

5.1 No Significant Hazards Consideration

Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., has evaluated the proposed 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.*

Part of the proposed changes introduces a Condition for an inoperable LPI with an AOT of seven days, introduces another Condition for an inoperable BS train coincident with an inoperable Containment Cooling train with an AOT of 72 hours, and extends the AOT for one inoperable BS train, DC train, and/or RW train to seven days. These systems are not initiators for any accident previously evaluated. The consequences of an event during the extended Completion Time are no more severe than the consequences of the same event during the current Completion Time. Therefore, the consequences of an event previously analyzed are not increased, so the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Another part of the proposed changes eliminates second Completion Times from the CR-3 ITS. Second Completion Times are not an initiator to any accident previously evaluated. As a result, the probability of an accident previously evaluated is not affected. The consequences of an accident during the revised Completion Time are no different from the consequences of the same accident during the existing Completion Times. As a result, the consequences of an accident previously evaluated are not affected by this change. The proposed changes do not alter or prevent the ability of SSCs from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences.

The proposed editorial/administrative changes remove obsolete information and provide clarification. These changes do not affect any system that is an initiator for any accidents previously evaluated. The consequences of an accident previously evaluated are not affected. The proposed changes do not alter or prevent the ability of SSCs from performing their intended function to mitigate the consequences of an initiating event. The proposed editorial/administrative changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed editorial/administrative changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences.

Therefore, the proposed changes do 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.*

The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. These changes do not alter any assumptions made in the safety analysis.

Therefore, 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 on safety.*

One part of the proposed changes introduces a Condition for an inoperable LPI with an AOT of seven days, introduces another Condition for an inoperable BS train coincident with an inoperable Containment Cooling train with an AOT of 72 hours, and extends the AOT for one inoperable BS train, DC train, and/or RW train to seven days. An evaluation presented in Reference 8.3, and accepted by the NRC, concluded that the extended Completion Time did not result in a significant reduction in the margin of safety. An analysis performed by FPC also drew the same conclusion. Therefore, extending the AOT to seven days for these components does not involve a significant reduction in a margin of safety.

The proposed change to delete the second Completion Time from the CR-3 ITS does not alter the manner in which safety limits, limiting safety system settings or LCOs are determined. The safety analysis acceptance criteria are not affected by this change. The proposed changes will not result in plant operation in a configuration outside of the design basis.

Similarly, the proposed editorial/administrative changes do not alter the manner in which safety limits, limiting safety system settings or LCOs are determined. The safety analysis acceptance criteria are not affected by this change. As such, the proposed editorial/administrative changes will not result in plant operation in a configuration outside of the design basis.

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

Based on the above, FPC concludes that the proposed changes to the CR-3 ITS present 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.

5.2 Applicable Regulatory Requirements / Criteria

The analysis in Reference 8.3 demonstrates the proposed AOT extension to seven days for LPI, BS, DC and RW Systems continue to meet the applicable regulatory requirements.

10 CFR 50.36, "Technical Specifications." 10 CFR 50.36(c)(2) states, "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall

shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." The revised Actions to eliminate second Completion Times limiting time from discovery of failure to meet an LCO continue to meet the requirements of this regulation.

10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The overall objective of this performance-based rule is to ensure that nuclear power plant SSCs will be maintained so that they will perform their intended function when required.

Based on the considerations discussed above for all requested changes, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an Operating License for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

FPC has reviewed proposed License Amendment Request #295, Revision 0, and concludes it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with this request.

7.0 PRECEDENTS

There are similarities between this LAR and an amendment granted to Duke Energy Corporation for Oconee Units 1, 2 and 3 (Docket Numbers 50-269, 50-270, and 50-287) on June 18, 2003, that extended the Completion Time for an inoperable LPI train from 72 hours to seven days (as presented in TSTF-430). A LAR similar to this one was submitted by Texas Utilities for Comanche Peak (Docket Numbers 50-445 and 50-446) on December 19, 2006, to eliminate second Completion Times from the TSs (as presented in TSTF-439). Finally, Amendment 220 was granted to CR-3 on September 6, 2005, that removed an obsolete footnote from ITS 3.7.9, Nuclear Services Seawater

System, and 3.8.1, AC Sources – Operating, similar to the footnotes proposed for deletion in this LAR.

8.0 REFERENCES

8.1 Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-430, Revision 2 (BWOG-104, Revision 1) dated December 22, 2003, "AOT Extension to 7 Days for LPI and Containment Spray (BAW-2295-A, Revision 1)"

8.2 Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-439-A, Revision 2 (WOG-165, Revision 0) dated December 19, 2005, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO"

8.3 BAW-2295-A, Revision 1, dated September 1999, "Justification for Extension of AOT for Low Pressure Injection and Reactor Building Spray Systems"

8.4 NRC Letter from T.H. Boyce (NRC) to Technical Specification Task Force dated August 5, 2004, Safety Evaluation by the Office of Nuclear Reactor Regulation TSTF-430, Implementation of B&W Topical BAW-2295, Revision 1 "Justification for Extension of AOT for Low Pressure Injection and Reactor Building Spray Systems" (TAC No. MA3939)

8.5 NRC Letter from T.H. Boyce (NRC) to Technical Specification Task Force dated January 11, 2006, Status of TSTF 439, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO"

8.6 Memorandum from Gordon Vytlacil (NRC) to TSPS (NRC) dated August 5, 1991, "Summary of Potential Allowed Outage Time (AOT) Extension Issue"

8.7 Gordon M. Vytlacil (NRC) to Lee Bush (WOG), et al dated December 16, 1991, "Information on the Completion Time Cap – to be discussed at Wednesdays meeting with Chris Grimes"

8.8 RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

8.9 RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"

8.10 Crystal River Unit 3 PRA, Florida Power Corporation, Science Applications Intl. Corporation, July 1987

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

ATTACHMENT B

Proposed Crystal River Unit 3 Operating License Changes

2.C.(9) Florida Power Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports, dated July 27, 1979, January 22, 1981, January 6, 1983, July 18, 1985, and March 16, 1988 subject to the following provisions:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. *{AMENDMENT NO. 147, dated January 22, 1993}*

2.C.(10) The design of the reactor coolant pump supports need not include consideration of the effects of postulated ruptures of the primary reactor coolant loop piping and may be revised in accordance with Florida Power Corporation's amendment request of April 24, 1986. *{AMENDMENT NO. 89, dated May 23, 1986}*

2.C.(11) A system of thermocouples added to the decay heat (DH) drop and Auxiliary Pressurizer Spray (APS) lines, capable of detecting flow initiation, shall be operable for Modes 4 through 1. Channel checks of the thermocouples shall be performed on a monthly basis to demonstrate operability. If either the DH or APS system thermocouples become inoperable, operability shall be restored within 30 days or the NRC shall be informed, in a Special Report within the following fourteen (14) days, of the inoperability and the plans to restore operability. *{AMENDMENT NO. 164, dated January 27, 1998}*

2.C.(12) Florida Power Corporation shall assure that the Cycle 14 core for CR-3 is designed using the methods specified in and operated within the Core Operating Limits Report limits developed from Topical Reports BAW-10164P-A, Revision 4, and BAW-10241P, Revision 0, in addition to those methods allowed by Improved Technical Specification 5.6.2.18. *{AMENDMENT NO. 211, dated October 16, 2003}*

~~2.C.(13)~~ ~~During the one-time extended allowed outage time for work on Nuclear Services Seawater System Emergency Pump RWP-2B, authorized in Improved Technical Specification (ITS) 3.7.9, CR-3 will implement the following compensatory measures:~~

> ~~(a)~~ ~~No elective maintenance will be scheduled on other related risk sensitive equipment beyond that required for the refurbishment activity of RWP-2B that could degrade the risk profile of the plant. For these purposes, the systems are: Nuclear Services and Decay Heat Seawater System, Decay Heat System, Decay Heat Closed Cycle Cooling Water System, Nuclear Services~~

~~Closed Cycle Cooling Water (SW), Emergency Diesel Generators, Chilled Water, Emergency Feedwater System, Emergency Feedwater Initiation and Controls System (EFIC) Auxiliary Feedwater Pump and Makeup System.~~

~~(b) Makeup Pump configurations will be selected to minimize risk.~~

~~(c) Increased operator attention will be focused on loss/restoration of RW/SW/ redundant train and the Appendix R Chiller. This will be accomplished by on shift operating crew review of Abnormal Procedure (AP)-330, Loss of Nuclear Service Cooling.~~

~~(d) Operator attention will be focused on potential use of non-safety grade Feedwater Pump FWP-7 and its dedicated Diesel Generator (MTDG-1). This will be accomplished by on shift operating crew review of Emergency Operating Procedure (EOP)-14, Enclosure 7, Emergency Feedwater Pump (EFWP) Management.~~

~~(e) Daily operator walkdowns of the redundant train of RW/SW/Pumps and associated power supply switchgear will be conducted.~~

~~(f) No elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES Buses or to the Bus for RWP-1.~~

~~(g) Hourly roving fire watches will be established in fire zones identified as containing circuits required for RWP-1 or RWP-2A to minimize fire risk in these areas. Those fire zones are: AB-95-3AA, AB-95-3B, AB-95-3E, AB-95-3F, AB-95-3G, AB-95-3K, AB-95-3T, AB-95-3U, AB-95-3W, AB-95-3X, AB-95-3Z, CC-108-102, CC-108-104, CC-108-105, CC-108-106, CC-108-108, CC-108-109, CC-108-110, CC-124-111, CC-124-116, CC-124-117, CC-134-118A, IB-95-200C, IB-119-201B, TB-119-400E, TB-119-403, TB-95-400A, TB-95-401, AB-119-6 and AB-119-6A.~~

~~(h) Prior to entering the Allowed Outage Time for rebuild of RWP-2B, fire zones containing circuits for either RWP-1 or RWP-2A will be walked down to identify and minimize transient combustibles not related to ongoing approved work. Hourly roving fire watches will be charged with continuing to monitor for the presence of transient combustible materials until RWP-2B is returned to service.~~

~~(i) For the risk significant fire zones containing circuits for both RWP-1 and RWP-2A, the following additional~~

~~compensatory measures will be established which address the specific risk factors in each zone. No additional compensatory measures beyond roving fire watches will be established for fire zones that contain circuits for all three pumps (RWP-1, RWP-2A and RWP-2B).~~

~~Fire Zone CC-108-108~~
~~A continuous Fire Brigade Qualified fire watch will be stationed in the fire area except while 4160V breaker manipulations are being performed. The individual's turnout gear will be available in the adjacent area where a fire hose station is also located equipped with an electrically safe fog nozzle.~~

~~Fire Zone AB-95-3W~~
~~The Waste Transfer Pumps will only be operated when there is a qualified fire watch in the immediate vicinity of the operating pump equipped with a radio and fire extinguisher. No hot or spark producing work will be conducted. A roving hourly fire watch will observe this zone.~~

~~Fire Zone AB-95-3E~~
~~The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.~~

~~Fire Zone AB-95-3AA~~
~~The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.~~

~~Fire Zone AB-95-3F~~
~~The operating makeup pump will be selected based on minimizing the risk from internal events as a result of redundancy in its cooling water source. No hot or spark producing work will be conducted. Roving hourly fire watches will be conducted.~~

~~Fire Zone AB-95-3T~~
~~The portion of the fire zone that can be locked will be locked and the keys will be administratively controlled. Entries will be limited to only operationally necessary activities and require inspection for transient combustible materials upon exit. The portion of the zone that cannot be locked will be observed by the roving fire watch. No hot or spark producing work will be conducted.~~

~~**Fire Zone AB-95-3U**~~
~~**The portion of the fire zone that can be locked will be locked and the keys will be administratively controlled. Entries will be limited to only operationally necessary activities and require inspection for transient combustible materials upon exit. The portion of the zone that cannot be locked will be observed by the roving fire watch. No hot or spark producing work will be conducted.**~~
~~*{AMENDMENT NO. 212, dated May 18, 2004]*~~

2.D **The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21 is entitled: "Crystal River Nuclear Power Plant Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0" submitted by letter dated September 30, 2004, as supplemented by letter dated October 20, 2004.** *{Revised by letter dated October 28, 2004]*

2.D.(3) **"Crystal River Nuclear Plant Unit 3 Security Training and Qualification Plan", Revision 3, dated December 30, 1981, submitted by letter dated March 19, 1982, and consisting of all previous revisions. This plan shall be followed in accordance with 10 CFR 73.55(b)(4), 60 days after approval by the Commission. All security personnel, as required in the above plans, shall be qualified within two years of this approval. The licensee may make changes to this plan without prior Commission approval if the changes do not decrease the safeguards effectiveness of the plan. The licensee shall maintain records of and submit reports concerning such changes in the same manner as required for changes made to the Security Plan and Safeguards Contingency Plan pursuant to 10 CFR 50.54(p).** *{AMENDMENT NO. 62, dated March 4, 1983}*

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

ATTACHMENT C

Proposed Improved Technical Specification and Bases Changes (Mark-up)

1.3 Completion Times

DESCRIPTION (continued)

However, when a subsequent train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." ~~Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Condition A and B in Example 1.3-3 may not be extended.~~

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

~~The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.~~

It is possible to alternate between Condition A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS-Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Low Pressure Injection (LPI) subsystem inoperable.	A.1 Restore LPI subsystem to OPERABLE status.	7 days
B~~A~~. One or more trains inoperable for reasons other than Condition A. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	B~~A~~.1 Restore train(s) to OPERABLE status.	72~~*~~ hours
C~~B~~. Required Action and associated Completion Time not met.	C~~B~~.1 Be in MODE 3. AND C~~B~~.2 Be in MODE 4.	6 hours 12 hours

~~*On a one-time basis, an Emergency Core Cooling System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

3.6 CONTAINMENT SYSTEMS

3.6.6 Reactor Building Spray and Containment Cooling Systems

LCO 3.6.6 Two reactor building spray trains and two containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One reactor building spray train inoperable.	A.1	Restore reactor building spray train to OPERABLE status.	~~72* hours~~ 7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
B. Required Action and associated Completion Time of Condition A not met.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 84 hours
C. One required containment cooling train inoperable.	C.1	Restore required containment cooling train to OPERABLE status.	7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~

(continued)

~~*On a one-time basis, a Reactor Building Spray System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One reactor building spray and one required containment cooling train inoperable.	D.1 Restore reactor building spray train to OPERABLE status. OR D.2 Restore required containment cooling train to OPERABLE status.	72 hours 72 hours
~~D~~E. Two required containment cooling trains inoperable.	~~D~~E.1 Restore one required containment cooling train to OPERABLE status.	72 hours
~~E~~F. Required Action and associated Completion Time of Condition ~~C or D~~ C, D, or E not met.	~~E~~F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours
~~F~~G. Two reactor building spray trains inoperable. OR Any combination of three required trains inoperable.	~~F~~G.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each reactor building spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to the turbine driven EFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~
B. One EFW train inoperable for reasons other than Condition A.	B.1 Restore EFW train to OPERABLE status.	72 hours ~~*~~ ~~AND~~ ~~10 days from discovery of failure to meet the LCO*~~

(continued)

~~*On a one-time basis, an EFW train may be inoperable for up to 14 days to allow performance of EFW Pump (EFP-3) repairs. The ability to apply the 14-day Completion Time will expire on March 31, 2005.~~

3.7 PLANT SYSTEMS

3.7.8 Decay Heat Closed Cycle Cooling Water (DC) System

LCO 3.7.8 Two DC trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC train inoperable.	A.1 ------NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by DC train inoperability. --------------------- Restore DC train to OPERABLE status.	~~72* hours~~ 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3	6 hours
	<u>AND</u>	
	B.2 Be in Mode 5.	36 hours

~~*On a one-time basis, a Decay Heat Closed Cycle Cooling Water System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

3.7 PLANT SYSTEMS

3.7.10 Decay Heat Seawater System

LCO 3.7.10 Two Decay Heat Seawater System trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Decay Heat Seawater System train inoperable.	A.1 --------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by Decay Heat Seawater System train inoperability. --------------------- Restore Decay Heat Seawater System train to OPERABLE status.	~~72* hours~~ 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3 AND B.2 Be in Mode 5.	6 hours 36 hours

~~*On a one-time basis, a Decay Heat Seawater System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status	72 hours ~~AND~~ ~~6 days from discovery of failure to meet LCO~~ ~~OR~~ ~~17 days if alternate AC power is available~~
B. One EDG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable EDG, inoperable when its redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Determine OPERABLE EDG is not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE EDG.	24 hours
	AND	
	B.4 Restore EDG to OPERABLE status	72 hours ~~AND~~ ~~6 days from discovery of failure to meet LCO~~ OR 14 days if alternate AC power is available ~~AND~~ ~~17 days from discovery of failure to meet LCO~~

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems-Operating

LCO 3.8.9 Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystem inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours ~~AND~~ ~~16 hours from discovery of failure to meet LCO~~
B. One AC vital bus subsystem inoperable.	B.1 Restore AC vital bus subsystem to OPERABLE status.	8 hours ~~AND~~ ~~16 hours from discovery of failure to meet LCO~~

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours ~~AND~~ ~~16 hours from discovery of failure to meet LCO~~
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Two trains with inoperable distribution subsystems that result in a loss of function.	E.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance is based on the small break LOCA, which establishes the pump performance curve and is less dependent on power. MODES 2 and 3 requirements are bounded by the MODE 1 analysis.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.6, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one LPI subsystem inoperable, action must be taken to restore it to OPERABLE status within 7 days. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPI subsystem could result in loss of ECCS function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable LPI subsystem. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 5. Reference 5 concluded that extending the Completion Time to 7 days for an inoperable LPI subsystem proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the LPI subsystem unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

(continued)

BASES

ACTIONS

B.1

With one or more ECCS trains inoperable and at least 100% of the flow equivalent to a single OPERABLE ECCS train available, the ~~inoperable~~ components inoperable for reasons other than Condition A must be returned to OPERABLE status within 72~~*~~ hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for many repairs.

~~*On a one-time basis, an Emergency Core Cooling System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that the safety injection (SI) flow equivalent to 100% of a single train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

For example, removing a train of the recirculation line to the RB sump or the entire bank of valves for maintenance does not render the HPI System inoperable, given the diverse ability to recirculate to the Makeup Tank. HPI satisfies Criterion 3 of the NRC Policy Statement which addresses SSCs that are part of the primary success path, and which function or actuate to mitigate a design basis accident or transient challenging a fission product barrier. Since this recirculation line supports piggyback operation in long-term cooling, and piggyback operation is not a primary success path, LCO 3.5.2 need not be entered when this recirculation path is not available. Similarly, Condition B does not have to be entered when an associated LPI train is unable to support HPI piggyback operation. The risk associated with this configuration was considered in the development of Condition A.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES

ACTIONS (continued)

~~B~~C.1 and ~~B~~C.2

If the inoperable components cannot be returned to OPERABLE status within the associated Completion Times, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and at least MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path. There are several exceptions for valve position verification due to the low potential for these types of valves to be mispositioned. The valve types which are not verified as part of this SR include vent or drain valves, relief valves, instrumentation valves, check valves, and sample line valves. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. For a power operated valve to be considered "locked, sealed, or otherwise secured", the component must be electrically and physically restrained. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Periodic inspections of the reactor building emergency sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and to preserve access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

REFERENCES

1. 10 CFR 50.46.
2. FSAR, Section 6.1.
3. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
4. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWP-3000.
5. ~~Deleted.~~ BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.
6. FSAR, Section 4.3.10.1.
7. Letter from NRC to FPC, 3N1098-15, dated October 29, 1998, "Issuance of Exemption from the Requirements of 10 CFR 50, Appendix K, Section I.D.1 - Crystal River Unit 3 (TAC No. M99892)".

BASES

LCO (continued)

iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two RB spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each RB Spray System train includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building emergency sump.

Each Containment Cooling System train includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the RB spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the RB Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

~~With one RB spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72* hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat~~

(continued)

BASES

ACTIONS

A.1 (continued)

~~removal capability afforded by the OPERABLE RB spray train and cooling system train(s), reasonable time for repairs, and the low probability of a DBA occurring during this period.~~

~~*On a one-time basis, a Reactor Building Spray System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

With one reactor building spray train inoperable, action must be taken to restore it to OPERABLE status within 7 days. In this condition, the remaining OPERABLE reactor building spray train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining reactor building spray train could result in loss of spray function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable reactor building spray train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 5. Reference 5 concluded that extending the Completion Time to 7 days for an inoperable reactor building spray train proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the reactor building spray train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

~~The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times", for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.~~

B.1 and B.2

If the inoperable RB spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner

(continued)

and without challenging plant systems. The extended interval to reach MODE 5 allows additional time to attempt restoration of the RB spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

C.1 (continued)

~~The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.~~

D.1 and D.2

With one reactor building spray and one required containment cooling train inoperable, one of the required inoperable trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Containment Cooling System, the iodine removal function of the Reactor Building Spray System, and the low probability of a DBA occurring during this period.

~~D~~E.1

With two of the required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition (both spray trains are OPERABLE or else Condition ~~E~~G is entered) provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

~~E~~F.1 and ~~E~~F.2

If the Required Actions and associated Completion Times of Condition ~~C or D~~ C, D, or E of this LCO are not met, the plant must be placed in a MODE in which the LCO does not

(continued)

BASES

apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~F~~G.1

With two RB spray trains or any combination of three or more RB spray and required containment cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.2 (continued)

occurring between surveillances and has been shown to be acceptable through operating experience.

It is preferable to run the fans in slow speed for this SR since this provides additional confidence the post-accident containment cooling train circuitry is OPERABLE.

SR 3.6.6.3

Verifying that each RB spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. ~~5~~6). Since the RB Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.4

Verifying an emergency design cooling water flow rate of ≥ 1780 gpm to each required containment cooling system heat exchangers (fan cooling coils) ensures the design flow rate assumed in the safety analysis is being achieved. The SR verifies that, with the SW System in the post-accident ES alignment, adequate flow is provided to the heat exchangers to remove the design basis reactor building heat load. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. While the heat exchangers can be aligned to the SW System during normal operations, other critical normal-running SW loads make it impractical to verify accident flow rate to the heat exchangers with the plant on-line. On an ES actuation, these normal-running loads are isolated and the SW flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

For activities, such as a valve repair/replacement, a visual inspection would be the preferred post-maintenance test since small debris in a localized area is the most likely concern. A smoke or air test would be appropriate following an event where a large amount of debris entered the system or water was actually discharged through the spray nozzles. For an inadvertent actuation of the Reactor Building Spray system, an air or smoke test should be performed at the next outage of sufficient duration.

REFERENCES

1. FSAR, Section 1.4.

2. FSAR, Section 14.2.2.5.9.

3. FSAR, Section 6.3.

4. RO-2787 Requirement Outline, Reactor Building Fan Assemblies, Addendum B, February 19, 1971.

5. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

~~5~~6. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

LCO (continued)

Inoperability of the EFW System may result in inadequate decay heat removal following a transient or accident during which main feedwater is not available. The resulting RCS heatup and pressure increase can potentially result in significant loss of coolant through the pressurizer code safety valves or the PORV.

APPLICABILITY

In MODES 1, 2, and 3 the EFW System is required to be OPERABLE and to function in the event that main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the secondary side inventory lost as the plant cools to MODE 4 conditions.

In MODES 4, 5 and 6, the OTSG need not be used to cooldown the RCS. Therefore, the EFW System is not required to be OPERABLE in these MODES.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable EFW train when entering MODE 1. There is an increased risk associated with entering MODE 1 with EFW inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, action must be taken to restore the steam supply to OPERABLE status within 7 days. Allowing 7 days in this Condition is reasonable, based on the redundant OPERABLE steam supply to the pump and the low probability of an event occurring that would require the inoperable steam supply to the turbine driven EFW pumps.

~~The 10 day Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be entered during any continuous failure to meet this LCO. The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The 'AND' connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.~~

(continued)

BASES

ACTIONS (continued)

B.1

If one of the EFW trains is inoperable, action must be taken to restore the train to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a DBA occurring during this time period. This condition includes the loss of two steam supply lines to the turbine driven EFW pump.

~~The 10 day Completion Time for Required Action B.1 established a limit on the maximum time allowed for any combination of Conditions to be entered during any continuous failure to meet this LCO. The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The 'AND' connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.~~

~~*On a one-time basis, an EFW train may be inoperable for up to 14 days to allow performance of EFW Pump (EFP-3) repairs. The ability to apply the 14-day Completion Time will expire on March 31, 2005.~~

C.1 and C.2

If Required Action A.1 or Required Action B.1 cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With both EFW trains inoperable, the plant is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, plant operation should not be perturbed by a forced action, including a power change, that might result in a trip. For this reason, the Technical Specifications do not mandate a plant shutdown. Rather the ACTIONS allow the plant to dictate the most prudent course of action (including plant shutdown) for the situation. The seriousness of this condition requires that action be initiated immediately to restore at least one EFW train to OPERABLE status.

(continued)

BASES

ACTIONS

A.1 (continued)

With one DC train inoperable, action must be taken to restore the train to OPERABLE status within ~~72* hours~~ 7 days. In this Condition, the remaining OPERABLE DC train is adequate to perform the heat removal function. The ~~72 hour~~ 7 day Completion Time for restoring full DC System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the DC System. ~~This Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.~~ The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable DC train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 3. Reference 3 concluded that extending the Completion Time to 7 days for an inoperable DC train proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the DC train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

~~*On a one-time basis, a Decay Heat Closed Cycle Cooling Water System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

B.1 and B.2

If the inoperable DC train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and power operated valves in the DC flow path provides assurance that the proper flow paths exist for DC operation. The isolation of the DC flow to individual components may render those components inoperable, but does not affect the operability of the DC system. This SR does not apply to

(continued)

BASES

valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

There are several exceptions for valve position verification due to the low potential for these types of valves to be mispositioned. The valve types which are not verified as part of this SR include vent or drain valves, relief valves, instrumentation valves, check valves, and sample line valves. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. For a power operated valve to be considered "locked, sealed, or otherwise secured," the component must be electrically and physically restrained. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in their correct position.

The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the DC pumps on an actual or simulated actuation signal. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was considered acceptable from a reliability standpoint.

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

REFERENCES

1. FSAR, Section 9.5.2.2.
2. Enhanced Design Basis Document for Decay Heat Closed Cycle Cooling Water System.
3. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops - MODE 4," should be entered if an inoperable decay heat seawater train results in an inoperable required DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for an inoperability of a required DHR loop.

If one of the decay heat seawater trains is inoperable, action must be taken to restore the train to OPERABLE status within ~~72* hours~~ 7 days. In this Condition, the remaining OPERABLE train is adequate to perform the heat removal function. The ~~72 hour~~ 7 day Completion Time for restoring full Decay Heat Seawater System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the Decay Heat Seawater System. ~~This Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.~~ The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable decay heat seawater train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 3. Reference 3 concluded that extending the Completion Time to 7 days for an inoperable decay heat seawater train proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the decay heat seawater train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

~~*On a one-time basis, a Decay Heat Seawater System train may be inoperable as specified by Required Action A.1 for up to 10 days to allow performance of Decay Heat Seawater System Pump RWP-3B repairs online. Upon completion of the refurbishment and system restoration this footnote is no longer applicable.~~

B.1 and B.2

If the inoperable decay heat seawater train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for manual valves in the Decay Heat Seawater System flow path provides assurance that the proper flow paths exist for DC operation. This SR does

(continued)

BASES

not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

There are several exceptions for valve position verification due to the low potential for these types of valves to be mispositioned. The valve types which are not verified as part of this SR include vent or drain valves, relief valves, instrumentation valves, check valves and sample line valves. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. For a power operated valve to be considered "locked, sealed, or otherwise secured," the component must be electrically and physically restrained. This surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.10.2

This SR verifies proper automatic operation of the decay heat seawater pumps on an actual or simulated actuation signal. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

REFERENCES

1. Enhanced Design Basis Document for Decay Heat Closed Cycle Cooling Water System.
2. FSAR, Section 9.5.2.2.
3. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

BASES

ACTIONS

A.2 (continued)

If at any time during the existence of Condition A (one offsite circuit inoperable) both 'a' and 'b' above become met, this Completion Time begins to be tracked.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E distribution system. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to the recommendations of Regulatory Guide 1.93 (Ref. 6), operation with one required offsite circuit inoperable should be limited to a period of time not to exceed 72 hours. In this condition, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. However, the remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E distribution system.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

~~The 6 day (17 days with the alternate AC source available) Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failure to meet the LCO. If Condition A is entered while, for instance, an EDG is inoperable and that EDG is subsequently returned to OPERABLE status, LCO 3.8.1 may already have been not met for up to 14 days. This could lead to a total of 17 days, since initial failure to meet the LCO, to restore the offsite circuit.~~

(continued)

BASES

ACTIONS

A.3 (continued)

~~The 6 day and 17 day Completion Times provide limits on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently.~~

~~As in Required Action A.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.~~

B.1

To ensure a highly reliable power source in the event one EDG is inoperable, it is necessary to verify the availability of the OPERABLE offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met (Condition F). However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a EDG is inoperable, does not result in a complete loss of safety function of critical redundant required features. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable EDG. Single train systems (from an electrical perspective), such as the turbine driven emergency feedwater pump, are not included.

(continued)

ACTIONS (continued)

B.4 (continued)

A periodic fire watch will be established in fire areas that are considered risk-significant by the IPEEE, affect both EDGs or have increased risk significance due to EDG maintenance. The fire areas are listed in Table B 3.8.1-1.

~~The 17-day Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failure to meet the LCO. Refer to the Bases for Required Action A.3 for additional information on this Completion Time.~~

(continued)

ACTIONS

A.1 (continued)

The most severe scenario addressed by Condition A is an entire train without AC power (i.e., no offsite power to the train and the associated EDG inoperable). In this condition, the plant has an increased vulnerability to a complete loss of AC power. It is, therefore, imperative that the operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the plant, and on restoring power to the affected train. The 8 hour time limit for restoration, prior to requiring a plant shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with shutting down the plant within this time limit; and

b. The low probability of an event occurring coincident with a single failure of a redundant component in the train with AC power.

~~The 16 hour Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failure to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored to OPERABLE status, LCO 3.8.9 may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored to OPERABLE status. This could continue indefinitely.~~

~~The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.~~

(continued)

ACTIONS

B.1 (continued)

The 8 hour Completion Time takes into account the importance of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

~~The 16 hour Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failure. Refer to the Bases for Required Action A.1 for further discussion of this Completion Time.~~

C.1

With DC bus(es) in DC electrical power distribution train inoperable, the remaining train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution train could result in the minimum required ES functions not being met. Therefore, the DC buses must be restored to OPERABLE status within 2 hours.

Condition C represents a condition in which one train is without adequate DC power; potentially both with the battery significantly degraded and the associated charger inoperable. In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

The 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without adequate AC vital power. However, there are certain affected features with Completion Times of shorter duration. The intent of the Improved Technical Specifications is to remain within this Specification only and not take the ACTIONS for inoperable supported systems. Taking this exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion

(continued)

BASES

ACTIONS

C.1 (continued)

Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and

c. The low probability of an event occurring coincident with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with the recommendations of Regulatory Guide 1.93 (Ref. 3).

~~The 16 hour Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failure to meet the LCO. Refer to the Bases for Required Action A for further discussion of this Completion Time.~~

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging plant systems.

(continued)

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

ATTACHMENT D

Proposed Improved Technical Specification and Bases Changes (Revision Bar Format)

1.3 Completion Times

DESCRIPTION (continued)

However, when a <u>subsequent</u> train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the <u>first</u> inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ."

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

It is possible to alternate between Condition A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS-Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Low Pressure Injection (LPI) subsystem inoperable.	A.1 Restore LPI subsystem to OPERABLE status.	7 days
B. One or more trains inoperable for reasons other than Condition A. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	B.1 Restore train(s) to OPERABLE status.	72 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Reactor Building Spray and Containment Cooling Systems

LCO 3.6.6 Two reactor building spray trains and two containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor building spray train inoperable.	A.1 Restore reactor building spray train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours
C. One required containment cooling train inoperable.	C.1 Restore required containment cooling train to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One reactor building spray and one required containment cooling train inoperable.	D.1	Restore reactor building spray train to OPERABLE status.	72 hours
	<u>OR</u>		
	D.2	Restore required containment cooling train to OPERABLE status.	72 hours
E. Two required containment cooling trains inoperable.	E.1	Restore one required containment cooling train to OPERABLE status.	72 hours
F. Required Action and associated Completion Time of Condition C, D, or E not met.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 5.	36 hours
G. Two reactor building spray trains inoperable. <u>OR</u> Any combination of three required trains inoperable.	G.1	Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each reactor building spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

LCO 3.7.5 Two EFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to the turbine driven EFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days
B. One EFW train inoperable for reasons other than Condition A.	B.1 Restore EFW train to OPERABLE status.	72 hours

(continued)

3.7 PLANT SYSTEMS

3.7.8 Decay Heat Closed Cycle Cooling Water (DC) System

LCO 3.7.8 Two DC trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC train inoperable.	A.1 ------NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by DC train inoperability. --------------------- Restore DC train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3	6 hours
	AND	
	B.2 Be in Mode 5.	36 hours

3.7 PLANT SYSTEMS

3.7.10 Decay Heat Seawater System

LCO 3.7.10 Two Decay Heat Seawater System trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Decay Heat Seawater System train inoperable.	A.1 --------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops-MODE 4," for required decay heat removal loops made inoperable by Decay Heat Seawater System train inoperability. --------------------- Restore Decay Heat Seawater System train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in Mode 3 <u>AND</u> B.2 Be in Mode 5.	6 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore required offsite circuit to OPERABLE status	72 hours
B. One EDG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable EDG, inoperable when its redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3.1	Determine OPERABLE EDG is not inoperable due to common cause failure.	24 hours
	<u>OR</u>		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE EDG.	24 hours
	<u>AND</u>		
	B.4	Restore EDG to OPERABLE status	72 hours <u>OR</u> 14 days if alternate AC power is available

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems-Operating

LCO 3.8.9 Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystem inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours
B. One AC vital bus subsystem inoperable.	B.1 Restore AC vital bus subsystem to OPERABLE status.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Two trains with inoperable distribution subsystems that result in a loss of function.	E.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance is based on the small break LOCA, which establishes the pump performance curve and is less dependent on power. MODES 2 and 3 requirements are bounded by the MODE 1 analysis.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.6, "RCS Loops–MODE 5, Loops Filled," and LCO 3.4.7, "RCS Loops–MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal and Coolant Circulation–High Water Level," and LCO 3.9.5, "Decay Heat Removal and Coolant Circulation–Low Water Level."

ACTIONS

A.1

With one LPI subsystem inoperable, action must be taken to restore it to OPERABLE status within 7 days. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPI subsystem could result in loss of ECCS function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable LPI subsystem. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 5. Reference 5 concluded that extending the Completion Time to 7 days for an inoperable LPI subsystem proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the LPI subsystem unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

(continued)

BASES

ACTIONS

B.1

With one or more ECCS trains inoperable and at least 100% of the flow equivalent to a single OPERABLE ECCS train available, the components inoperable for reasons other than Condition A must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for many repairs.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that the safety injection (SI) flow equivalent to 100% of a single train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

For example, removing a train of the recirculation line to the RB sump or the entire bank of valves for maintenance does not render the HPI System inoperable, given the diverse ability to recirculate to the Makeup Tank. HPI satisfies Criterion 3 of the NRC Policy Statement which addresses SSCs that are part of the primary success path, and which function or actuate to mitigate a design basis accident or transient challenging a fission product barrier. Since this recirculation line supports piggyback operation in long-term cooling, and piggyback operation is not a primary success path, LCO 3.5.2 need not be entered when this recirculation path is not available. Similarly, Condition B does not have to be entered when an associated LPI train is unable to support HPI piggyback operation. The risk associated with this configuration was considered in the development of Condition A.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the inoperable components cannot be returned to OPERABLE status within the associated Completion Times, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and at least MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path. There are several exceptions for valve position verification due to the low potential for these types of valves to be mispositioned. The valve types which are not verified as part of this SR include vent or drain valves, relief valves, instrumentation valves, check valves, and sample line valves. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. For a power operated valve to be considered "locked, sealed, or otherwise secured", the component must be electrically and physically restrained. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Periodic inspections of the reactor building emergency sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and to preserve access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and has been confirmed by operating experience.

REFERENCES

1. 10 CFR 50.46.
2. FSAR, Section 6.1.
3. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
4. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWP-3000.
5. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.
6. FSAR, Section 4.3.10.1.
7. Letter from NRC to FPC, 3N1098-15, dated October 29, 1998, "Issuance of Exemption from the Requirements of 10 CFR 50, Appendix K, Section I.D.1 - Crystal River Unit 3 (TAC No. M99892)".

BASES

LCO (continued)

iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two RB spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each RB Spray System train includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building emergency sump.

Each Containment Cooling System train includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the RB spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the RB Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one reactor building spray train inoperable, action must be taken to restore it to OPERABLE status within 7 days. In this condition, the remaining OPERABLE reactor building spray train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining reactor building spray train could result in loss of spray

(continued)

BASES

ACTIONS

A.1 (continued)

function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable reactor building spray train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 5. Reference 5 concluded that extending the Completion Time to 7 days for an inoperable reactor building spray train proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the reactor building spray train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

B.1 and B.2

If the inoperable RB spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time to attempt restoration of the RB spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

(continued)

ACTIONS (continued)

D.1 and D.2

With one reactor building spray and one required containment cooling train inoperable, one of the required inoperable trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Containment Cooling System, the iodine removal function of the Reactor Building Spray System, and the low probability of a DBA occurring during this period.

E.1

With two of the required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition (both spray trains are OPERABLE or else Condition G is entered) provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

F.1 and F.2

If the Required Actions and associated Completion Times of Condition C, D, or E of this LCO are not met, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

G.1

With two RB spray trains or any combination of three or more RB spray and required containment cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.6.2 (continued)

occurring between surveillances and has been shown to be acceptable through operating experience.

It is preferable to run the fans in slow speed for this SR since this provides additional confidence the post-accident containment cooling train circuitry is OPERABLE.

SR 3.6.6.3

Verifying that each RB spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 6). Since the RB Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.4

Verifying an emergency design cooling water flow rate of ≥ 1780 gpm to each required containment cooling system heat exchangers (fan cooling coils) ensures the design flow rate assumed in the safety analysis is being achieved. The SR verifies that, with the SW System in the post-accident ES alignment, adequate flow is provided to the heat exchangers to remove the design basis reactor building heat load. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. While the heat exchangers can be aligned to the SW System during normal operations, other critical normal-running SW loads make it impractical to verify accident flow rate to the heat exchangers with the plant on-line. On an ES actuation, these normal-running loads are isolated and the SW flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

For activities, such as a valve repair/replacement, a visual inspection would be the preferred post-maintenance test since small debris in a localized area is the most likely concern. A smoke or air test would be appropriate following an event where a large amount of debris entered the system or water was actually discharged through the spray nozzles. For an inadvertent actuation of the Reactor Building Spray system, an air or smoke test should be performed at the next outage of sufficient duration.

REFERENCES

1. FSAR, Section 1.4.
2. FSAR, Section 14.2.2.5.9.
3. FSAR, Section 6.3.
4. RO-2787 Requirement Outline, Reactor Building Fan Assemblies, Addendum B, February 19, 1971.
5. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.
6. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

LCO (continued)

Inoperability of the EFW System may result in inadequate decay heat removal following a transient or accident during which main feedwater is not available. The resulting RCS heatup and pressure increase can potentially result in significant loss of coolant through the pressurizer code safety valves or the PORV.

APPLICABILITY

In MODES 1, 2, and 3 the EFW System is required to be OPERABLE and to function in the event that main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the secondary side inventory lost as the plant cools to MODE 4 conditions.

In MODES 4, 5 and 6, the OTSG need not be used to cooldown the RCS. Therefore, the EFW System is not required to be OPERABLE in these MODES.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable EFW train when entering MODE 1. There is an increased risk associated with entering MODE 1 with EFW inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With one of the two steam supplies to the turbine driven EFW pump inoperable, action must be taken to restore the steam supply to OPERABLE status within 7 days. Allowing 7 days in this Condition is reasonable, based on the redundant OPERABLE steam supply to the pump and the low probability of an event occurring that would require the inoperable steam supply to the turbine driven EFW pumps.

(continued)

BASES

ACTIONS (continued)

B.1

If one of the EFW trains is inoperable, action must be taken to restore the train to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a DBA occurring during this time period. This condition includes the loss of two steam supply lines to the turbine driven EFW pump.

C.1 and C.2

If Required Action A.1 or Required Action B.1 cannot be completed within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With both EFW trains inoperable, the plant is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, plant operation should not be perturbed by a forced action, including a power change, that might result in a trip. For this reason, the Technical Specifications do not mandate a plant shutdown. Rather the ACTIONS allow the plant to dictate the most prudent course of action (including plant shutdown) for the situation. The seriousness of this condition requires that action be initiated immediately to restore at least one EFW train to OPERABLE status.

(continued)

BASES

ACTIONS

A.1 (continued)

With one DC train inoperable, action must be taken to restore the train to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE DC train is adequate to perform the heat removal function. The 7 day Completion Time for restoring full DC System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the DC System. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable DC train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 3. Reference 3 concluded that extending the Completion Time to 7 days for an inoperable DC train proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the DC train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

B.1 and B.2

If the inoperable DC train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment for manual and power operated valves in the DC flow path provides assurance that the proper flow paths exist for DC operation. The isolation of the DC flow to individual components may render those components inoperable, but does not affect the operability of the DC system. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

There are several exceptions for valve position verification due to the low potential for these types of valves to be mispositioned. The valve types which are not verified as part of this SR include vent or drain valves, relief valves, instrumentation valves, check valves, and sample line valves. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. For a power operated valve to be considered "locked, sealed, or otherwise secured," the component must be electrically and physically restrained. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in their correct position.

The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the DC pumps on an actual or simulated actuation signal. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was considered acceptable from a reliability standpoint.

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

REFERENCES

1. FSAR, Section 9.5.2.2.
2. Enhanced Design Basis Document for Decay Heat Closed Cycle Cooling Water System.
3. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.5, "RCS Loops - MODE 4," should be entered if an inoperable decay heat seawater train results in an inoperable required DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for an inoperability of a required DHR loop.

If one of the decay heat seawater trains is inoperable, action must be taken to restore the train to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE train is adequate to perform the heat removal function. The 7 day Completion Time for restoring full Decay Heat Seawater System OPERABILITY is the same as that for the ECCS Systems, whose safety functions are supported by the Decay Heat Seawater System. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable decay heat seawater train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 3. Reference 3 concluded that extending the Completion Time to 7 days for an inoperable decay heat seawater train proves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the decay heat seawater train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

B.1 and B.2

If the inoperable decay heat seawater train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5. within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for manual valves in the Decay Heat Seawater System flow path provides assurance that the proper flow paths exist for DC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing.

These valves include valves in the main flow paths and the first normally closed valve in a branch line. In lieu of the first normally closed valve in the branch line, credit may be taken for verifying valve position of another valve downstream, providing the isolation of the flow path is achieved. Verifying correct valve alignment of valves immediately downstream of an unsecured valve still assures isolation of the flow path.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

There are several exceptions for valve position verification due to the low potential for these types of valves to be mispositioned. The valve types which are not verified as part of this SR include vent or drain valves, relief valves, instrumentation valves, check valves and sample line valves. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. For a power operated valve to be considered "locked, sealed, or otherwise secured," the component must be electrically and physically restrained. This surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.10.2

This SR verifies proper automatic operation of the decay heat seawater pumps on an actual or simulated actuation signal. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

REFERENCES

1. Enhanced Design Basis Document for Decay Heat Closed Cycle Cooling Water System.

2. FSAR, Section 9.5.2.2.

3. BAW-2295-A, Revision 1, Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems.

BASES

ACTIONS

A.2 (continued)

If at any time during the existence of Condition A (one offsite circuit inoperable) both 'a' and 'b' above become met, this Completion Time begins to be tracked.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E distribution system. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to the recommendations of Regulatory Guide 1.93 (Ref. 6), operation with one required offsite circuit inoperable should be limited to a period of time not to exceed 72 hours. In this condition, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. However, the remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E distribution system.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1

To ensure a highly reliable power source in the event one EDG is inoperable, it is necessary to verify the availability of the OPERABLE offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met (Condition F). However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a EDG is inoperable, does not result in a complete loss of safety function of critical redundant required features. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable EDG. Single train systems (from an electrical perspective), such as the turbine driven emergency feedwater pump, are not included.

(continued)

BASES

ACTIONS (continued)

B.4 (continued)

A periodic fire watch will be established in fire areas that are considered risk-significant by the IPEEE, affect both EDGs or have increased risk significance due to EDG maintenance. The fire areas are listed in Table B 3.8.1-1.

(continued)

BASES

ACTIONS

A.1 (continued)

The most severe scenario addressed by Condition A is an entire train without AC power (i.e., no offsite power to the train and the associated EDG inoperable). In this condition, the plant has an increased vulnerability to a complete loss of AC power. It is, therefore, imperative that the operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the plant, and on restoring power to the affected train. The 8 hour time limit for restoration, prior to requiring a plant shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with shutting down the plant within this time limit; and

b. The low probability of an event occurring coincident with a single failure of a redundant component in the train with AC power.

(continued)

BASES

ACTIONS

B.1 (continued)

The 8 hour Completion Time takes into account the importance of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

C.1

With DC bus(es) in DC electrical power distribution train inoperable, the remaining train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution train could result in the minimum required ES functions not being met. Therefore, the DC buses must be restored to OPERABLE status within 2 hours.

Condition C represents a condition in which one train is without adequate DC power; potentially both with the battery significantly degraded and the associated charger inoperable. In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

The 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without adequate AC vital power. However, there are certain affected features with Completion Times of shorter duration. The intent of the Improved Technical Specifications is to remain within this Specification only and not take the ACTIONS for inoperable supported systems. Taking this exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion

(continued)

BASES

ACTIONS

C.1 (continued)

Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and

c. The low probability of an event occurring coincident with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with the recommendations of Regulatory Guide 1.93 (Ref. 3).

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging plant systems.

(continued)

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

ATTACHMENT E

Low Pressure Injection and Reactor Building Spray AOT Extension Risk Assessment

SYSTEM #	N/A
CALC. SUB-TYPE	N/A
PRIORITY CODE	4
QUALITY CLASS	Non-safety

NUCLEAR GENERATION GROUP

P07-0001

(CALCULATION #)

Low Pressure Injection and Reactor Building Spray AOT Extension Risk Assessment

(Title including structures, systems, components)

☐ BNP UNIT _____

☒ CR3 ☐ HNP ☐ RNP ☐ NES ☐ ALL

APPROVAL Signed Electronically ☒

REV	PREPARED BY	REVIEWED BY	SUPERVISOR
0	Signature	Signature	Signature
	Name Scott A. Brinkman	Name David N. Miskiewicz	Name Robert Rishel
	Date 4/9/2007	Date 4/9/2007	Date 4/11/2007

(For Vendor Calculations)

Vendor__________________________ Vendor Document No. ______________

Owner's Review By _________________________ Date ________________

LIST OF EFFECTIVE PAGES

PAGE	REV	PAGE	REV	ATTACHMENTS		
i-v	0			Number	Rev	Number of Pages
1-27	0				0	
					0	
					0	
					0	
					0	
					0	
				AMENDMENTS		
				Letter	Rev	Number of Pages

Rev. #	Revision Summary (list of ECs incorporated)
0	This is the original issue of this calculation.

Document Indexing Table

Document Type (e.g. CALC, DWG, TAG, PROCEDURE, SOFTWARE)	**ID Number** (e.g., Calc No., Dwg. No., Equip. Tag No., Procedure No., Software name and version)	**Function** (i.e. IN for design inputs or references; OUT for affected documents)	**Relationship to Calc.** (e.g. design input, assumption basis, reference, document affected by results)	**Action** (specify if Doc. Services or Config. Mgt. to Add, Deleted or Retain) (e.g., CM Add, DS Delete)
CALC	P-02-0001, Rev.3	IN	Design input, Reference	Add
CALC	P-05-0001, Rev. 1	IN	Reference	Add

Record of Lead Review

Design P07-0001 **Revision** 0

The signature below of the Lead Reviewer records that:

- the review indicated below has been performed by the Lead Reviewer;
- appropriate reviews were performed and errors/deficiencies (for all reviews performed) have been resolved and these records are included in the design package;
- the review was performed in accordance with EGR-NGGC-0003.

☐ **Design Verification Review** ☒ **Engineering Review** ☐ **Owner's Review**

☐ Design Review
☐ Alternate Calculation
☐ Qualification Testing

☐ **Special Engineering Review** ______________________

☐ YES ☐ N/A **Other Records are attached**

David N. Miskiewicz	Signed Electronically	PSA	4/11/2007
Lead Reviewer (print)	**(sign)**	**Discipline**	**Date**

Item No.	Deficiency	Resolution
1	Add EGDG-1C discussion to section 3.1	done
2	Include alignment factors in sensitivity analysis	done
3	Address concurrent maintenance unavailability of RWP-1,2A,2B due to RW pump suction bay impact	done

FORM EGR-NGGC-0003-2-5
This form is a QA Record when completed and included with a completed design package. Owner's Reviews may be processed as stand alone QA records when Owner's Review is completed

Record of Interdisciplinary Reviews

PART I — DESIGN ASSUMPTION / INPUT REVIEW: APPLICABLE ☒ **Yes** ☐ **No**

The following organizations have reviewed and concur with the design assumptions and inputs used in this calculation:

Organization	Name	Signature	Date
Systems Engineering	Philip Saltsman	Signed Electronically	4/10/07
Systems Engineering	James Lane	Signed Electronically	4/9/07
Licensing	David Rothrock	Signed Electronically	4/9/07
Outage & Scheduling			

PART II — RESULTS REVIEW:

The following organizations are aware of the impact of the results of this calculation (on designs, programs and procedures):

Systems Engineering
☐ Yes ☒ NO

Name	Signature	Date

Comments:

Operations
☐ Yes ☒ NO

Name	Signature	Date

Comments:

Licensing

Name	Signature	Date
David Rothrock	Signed Electronically	4/9/07

Comments:

Outage & Scheduling

Name	Signature	Date

Comments:

TABLE OF CONTENTS

1.0 Purpose

This analysis calculates the quantitative impact of a proposed permanent risk informed Technical Specification change in the Allowable Outage Time (AOT) for the Low Pressure Injection, Reactor Building Spray, Decay Heat Closed Cycle Cooling and Decay Heat Seawater systems from 72 hours to 7 days (168 hours).

This is a plant-specific evaluation using the Crystal River Unit 3 (CR3) PSA model for online operation (ref. 1). The assessment follows the guidance set forth in NRC Regulatory Guide 1.174 (ref. 2) and Regulatory Guide 1.177 (ref. 6). It evaluates the changes in CDF and LERF, and provides suggested compensatory actions to minimize risk impacts of the proposed maintenance.

2.0 References

1. CR3 calculation P-02-0001, Rev.3, "CR3 PSA - Model of Record", March 2006
2. RG 1.174, "An Approach for Using PRA in Risk Informed Decisions on plant Specific Changes to the Licensing Basis"
3. CR3 IPEEE, Rev.1, March 1997
4. CR3 Improved Technical Specifications
5. "Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems" B&W Owner's Group Topical Report BAW-2295A, rev. 1, September 1999.
6. RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"
7. CR3 calculation P-05-0001, Rev. 1, "PSA Risk Assessment of RWP-3B Extended AOT", June 2005.

3.0 Design Inputs

The primary input for this analysis is the CR3 PSA Model of Record (ref. 1).

3.1 PRA Quality

The PSA model used in this calculation is the "CR-3 PSA Model of Record – MOR06." The base case Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) values are given below:

- CDF = 4.99×10^{-6} per year
- LERF = 3.69×10^{-7} per year

These numbers are generally lower than the industry average. A review of the results and the model has provided some reasons for a low CDF and LERF:

- Byron Jackson N-9000 Reactor Coolant Pump (RCP) seals are installed and are assumed to maintain their integrity as long as they have seal injection, or seal cooling, or the RCPs are tripped. This greatly reduces the likelihood of an RCP seal failure causing a Loss-of-Coolant Accident (LOCA).
- Offsite power is supplied from a 230 kV switchyard that has feeds from the grid and from three fossil plants onsite. CR-3 outputs to a separate 500 kV switchyard. Based on this, dependent loss of offsite power events occurring due to trip initiators is not considered a credible event.
- CR-3 has a third non-safety related diesel that can power an ES bus that adds additional redundancy for loss of offsite power scenarios.
- CR-3 maintains a diverse secondary cooling capability, including automatically actuated steam and diesel driven emergency feedwater pumps, a backup motor driven pump powered from the Engineered Safeguards (ES) bus, and a backup motor driven pump that is powered from normal offsite power or the alternate emergency diesel generator.
- CR-3 has three high head injection/makeup pumps each capable of providing adequate primary cooling via the pressurizer power-operated relief valve or pressurizer safeties at full Reactor Coolant System (RCS) pressure. The High Pressure Injection (HPI) pumps also have diverse support systems. Two of the pumps have backup cooling and one can be powered from either ES 4160 kV bus.
- CR-3 has separate safety-related service water systems for the decay heat removal system and nuclear services support for other systems. The nuclear services system also has a third non-safety related train that can cool normal loads.
- CR-3 has a dedicated chiller installed for 10CFR50 Appendix R (fire) considerations that is not dependent on service water.

The PSA inputs used for this application were generated using updated Individual Plant Examination (IPE) models developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and associated supplements. The original development work was a level one Probabilistic Risk Assessment (PRA) study completed in 1987 (Crystal River Unit 3 Probabilistic Risk Assessment, Florida Power Corporation, Science Applications Intl. Corporation, July 1987), which was submitted to the NRC and reviewed by Argonne National Laboratory (NUREG/CR- 5245). This study was subsequently updated for the Generic Letter 88-20 IPE submittal to include a level two containment analysis and an internal flooding analysis. The study was subject to reviews by the relevant CR-3 system engineers, and review of the event sequence analysis, quantification, and recovery analysis by the Nuclear Safety Supervisor at CR-3, a former Senior Reactor Operator.

Revisions to the models have been made to maintain the models consistent with plant design changes and operational changes. These changes have been made by individuals knowledgeable in risk assessment techniques and methods, and reviewed by plant Engineering and Operations personnel familiar with the plant design and operation. The current PSA model and the risk assessment performed for this application have been documented as a calculation.

Current administrative controls include written procedures and review of all model changes, data updates, and risk assessments performed using PSA methods and models. Risk assessments are performed by a PSA engineer, reviewed by another PSA engineer, and approved by the PSA Supervisor or designee. Procedures, PSA model documentation, and associated records for applications of the PSA models, are controlled documents.

Since the submittal of the original PRA study in 1987, the PSA models have been maintained consistent with the current plant configuration such that they are considered "living" models which reasonably reflect the as-build, as-operated plant. The PSA models are updated for different reasons, including plant changes and modifications, procedure changes, accrual of new plant data, discovery of modeling errors, and advances in PSA technology. The update process ensures that the applicable changes are implemented and documented in a timely manner so that risk analyses performed in support of plant operations reflect the current plant configuration, operating philosophy, and transient and component failure history. The PSA maintenance and update process is described in administrative procedure ADM-NGGC-0004, "Updates to PSA Models." Guidance to determine the need for a model update is provided in the procedure. Each PRA model update is documented in accordance with plant procedures governing the preparation of engineering calculations, which includes an independent review of each calculation that is an input to the PRA model of record.

PSA Software

Computer programs that process PSA model inputs are verified and validated in accordance with administrative procedure CSP-NGGC-2505, "Software Quality Assurance and Configuration Control of Business Computer Systems." This procedure provides for software verification and validation to ensure the software meets the software requirement specifications and functional requirements, and typically includes a comparison of results generated to the results generated from previously approved software.

Model Changes Since Submittal of the IPE

Since the submittal of the IPE, there have been several significant plant design changes incorporated into the PSA model that have resulted in a reduction in the core damage frequency. A summary of significant model changes incorporated due to these plant changes includes the following:

- BEST added ("A" and "B" safeguards trains powered from separate transformers)
- FWP-7 with alternate emergency diesel generator 1C installed
- Appendix R chiller installed
- EFP-3 installed

- Installed Alternate AC diesel, which can power an Essential Bus
- Low pressure injection BWST suction valves changed to normally open
- High pressure injection discharge throttle valves and cross-ties added
- Revision of emergency operating procedures reflected in human action probabilities

In addition to these plant changes, updates have been made to plant-specific data (through 1999) and initiating events data, as well as updates to the methods used for human reliability, common cause, internal flooding, and level two analyses.

As of the date of this submittal, there are no outstanding plant changes which would require a change to the PSA model, and no planned plant changes which would be implemented prior to the fall 2007 refueling outage which would require a change to the PSA model.

PSA Reviews

As discussed above, the original CR-3 PRA study was reviewed by Argonne National Laboratory as documented in NREG/CR-5245. For the IPE submittal, multiple levels of review were used, including an assessment by Engineering and Operations personnel familiar with the plant design and operation. Subsequent revisions to the PSA models were performed by qualified individuals with knowledge of PSA methods and plant systems. Involvement by Engineering and Operations personnel in providing input and review of results was obtained when required based on the scope of the changes being implemented.

The CR-3 PSA model and documentation was subjected to the industry peer certification review process in September 2001. The industry peer certification review was conducted by a diverse group of PSA engineers from other Babcock & Wilcox (B&W) plants, industry PSA consultants familiar with the B&W plant design, and a representative from the Institute of Nuclear Power Operations (INPO). The reviewers involved in the peer review process were not employees of the company, and were not involved with the development of the PRA model. The certification review covered all aspects of the PSA model and the administrative processes used to maintain and update the model. This review generated specific recommendations for model changes, as well as guidance for improvements to processes and methodologies used in the CR-3 PSA model, and enhancements to the documentation of the model and the administrative procedures used for model updates.

Following completion of this review, the CR-3 PSA model was revised to address each issue identified which affected the model. The significant changes identified included:

- Update of the plant-specific thermal-hydraulic analyses that provides the bases for accident sequences, system success criteria, and timing for operator actions.
- Revision of accident sequence logic for steam generator tube rupture (SGTR) and anticipated transient without scram (ATWS) mitigation.
- Development of an initiating event to address the loss of all raw water pumps (loss of ultimate heat sink).
- Update of the interfacing systems loss of coolant accident (ISLOCA) analyses.

- Update of the human reliability analysis including the dependency analysis for multiple operator action responses to an event, and
- Update of the level two analysis.

All peer review items which impact the PSA model have been addressed and are reflected in the PSA model used in this submittal.

At the time of the peer review, the level two model was not yet completed, and only a preliminary draft version, along with the original IPE level two results, were available for review. The level two model is now complete, and the findings identified from the peer certification review of the preliminary results and the IPE model have been addressed.

4.0 Assumptions

4.1 General

1. The maximum allowed AOT times for the Low Pressure Injection system, The Reactor Building Sprays, Decay Heat Closed Cycle Cooling and Decay Heat Seawater systems AOT will also be assumed to be 7 days.
2. The extended AOT will be used for each train once per operating year on average.
3. The extended AOT will not be performed on the A and B trains simultaneously.
4. The extended AOT occurs when the plant is in operating mode 1.
5. It is assumed that the AOT maintenance occurs simultaneously on the LPI, DHCCC, RW and RBS systems.
6. During performance of the AOT, the corresponding opposite train equipment and diesel are considered to be protected.
7. Shutdown risk was not considered in the delta risk evaluation, to ensure results are bounding. Because a direct comparison of online to offline risk is difficult to quantify, the shutdown risk of having a LPI train unavailable is assumed to be zero.
8. All other limiting conditions will remain unchanged.
9. External events can be evaluated qualitatively.

4.2 At Power default configuration baseline (same as MOR)

1. The plant is operating at 100% power
2. MUP-1B is running to provide normal RCS make-up
3. MUP-1B is ES selected for HPI and powered from the "A" bus
4. MUP-1C is ES selected for HPI and aligned to the DC system for cooling (Train B)
5. MUP-1A is aligned to the SW system for cooling and is not ES selected.
6. All MU system unavailability is applied to unselected MUP
7. RWP-1 and SWP-1C are operating to provide normal SW cooling.
8. "A" train HVAC equipment is running (CHHE-1A, CHP-1A, AHF-17A, AHF-19A, AHF-54A)
9. ES 4160V "A" bus is powered from the offsite power transformer (OPT).
10. ES 4160V "B" bus is powered from the backup ES transformer (BEST).
11. Unit power is provided from the auxiliary transformer.

5.0 Calculation/Analysis Details

5.1 Software

The following table contains a listing of the software used to perform this analysis.

Table 1 – PRA Software/Tools

Name	*Version*	*Description*
CAFTA	5.2	Manages PRA fault trees, databases, and results. It can also be used as a front end for quantification.
EOOS	3.4	Front end for performing PRA quantification and importance/risk analyses.
FORTE	3.0b	Engine for performing PRA quantification.
QRECOVER	2.3	Engine for applying rule based recoveries to cutsets.

5.2 Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

The baseline CDF and LERF for this analysis is taken from reference 1 as 4.99E-06/year, and 3.69E-07/year. They are based on solving the model using EOOS as the front end and a truncation of 1E-12 (CDF) or 1E-11 (LERF), using the default configuration as described in section 4.2.

5.3 Change in CDF (ΔCDF)

The requested extension in the AOT completion time is 7 days. Using a full 7 day interval for calculating ΔCDF would result in an overly-conservative estimate. The actual yearly average increase unavailability would be less. The delta increase in the allowed outage time per train is four days, therefore to calculate the ΔCDF four days of unavailability was added for each train.. This estimate is based on an assumed 7 day AOT being performed for each train once a year.

The annual unavailability of LPI/RBS/DHCCC/RW is assumed to increase by 96 hours per train. The current unavailability used in the PRA for each of these systems is listed below, along with the addition of an unavailability term that represents the extended AOT. The MOR06 fault tree was modified such that each existing maintenance term for the systems affected by the AOT was ORed with a term representing the additional 96 hours assumed incurred by the extended AOT. In addition an additional mutually exclusive pair was added to the model to account for the fact that the AOT will not be performed on the "A" train and "B" train systems simultaneously.

The assumption is made in this analysis that the extended AOT maintenance is performed simultaneously for the LPI, RBS, DHCCW, and RW systems.

Table 2 – Unavailability Values

Basic Event	Description	Value	Event ORed with original BE	Value
LPM001AM	DHP-1A TRAIN IN MAINTENANCE	1.03E-2	LPIAOTAM	1.1E-2
IPMBSPAM	RB SPRAY TRAIN A IN MAINTENANCE	9.4E-3		
SPMRW3AM	RWP-3A IN MAINTENANCE	8.6E-3		
SPMDHCAM	DHCCC TRAIN A IN MAINTENANCE	4E-3		
LPM001BM	DHP-1B TRAIN IN MAINTENANCE	1.03E-2	LPIAOTBM	1.1E-2
IPMBSPBM	RB SPRAY TRAIN B IN MAINTENANCE	9.4E-3		
SPMRW3BM	RWP-3B IN MAINTENANCE	8.6E-3		
SPMDHCBM	DHCCC TRAIN B IN MAINTENANCE	4E-3		

When the AOT maintenance terms are included, the CDF becomes 5.54E-6/yr

This represents a potential increase in CDF (ΔCDF) for the year of:

$$\Delta CDF = 5.54E\text{-}06/yr - 4.99E\text{-}06/yr = \mathbf{5.5E\text{-}07/yr}$$

Per Reg. Guide 1.174 changes in CDF less than 1E-6 are considered very small. It should be noted these results do not consider the offset of risk incurred during shutdown operation. Neglecting shutdown risk results in conservative results.

5.4 Change in LERF (ΔLERF)

Similar to the impact on CDF, the impact to LERF is another metric to be evaluated. The baseline LERF from reference 1 is 3.69E-07/yr. When the AOT maintenance terms are included the annual LERF increases to 3.70E-07/yr.

This represents a potential increase in LERF (ΔLERF) for the year of:

$$\Delta LERF = 3.70E\text{-}07/yr - 3.69E\text{-}07/yr = \mathbf{1E\text{-}09/yr}$$

Per Reg. Guide 1.174 changes in LERF less than 1E-7 are considered very small. It should be noted these results do not consider the offset of risk incurred during shutdown operation. Neglecting shutdown risk results in conservative results.

5.5 Incremental Conditional Core Damage Probability (ICCDP)

Another approach to evaluating the risk of equipment out of service (OOS) is to assess the peak CDF when the component is OOS, then subtracting the base CDF from this result and multiplying by the time the component is out of service to get the Incremental Conditional Core Damage Probability.

Solving the model with the "A" train AOT maintenance event set to 1.0 results in a CDF of 2.05E-05/year.

ICCDP is equal to the integration of the change in risk over time. For this assessment it can be determined as the increase in risk (ΔCDF) when the AOT is occurring multiplied by the duration of the AOT.

For train A assumed out of service the result is:

$$\Delta CDF = CDF_{AOT} - CDF_{BASE} = 2.05E\text{-}05/yr - 4.99E\text{-}06/yr = \mathbf{1.55E\text{-}05/yr}$$

$$ICCDP = \Delta CDF * AOT = 1.55E\text{-}05/yr * 168\ hours * (1yr/8760\ hours) = \mathbf{2.97E\text{-}07}$$

The A train ICCDP case assumes that the opposite train diesel (train B) is not out of service for maintenance since this is precluded by technical specifications. The analysis assumes that maintenance on High Pressure Injection (HPI), Emergency Feedwater (EFW), Auxiliary Feedwater (AFW), Emergency Feedwater Initiation and Control (EFIC), and Appendix R chiller (CHP-2) will not be performed concurrently with the extended AOT. It also assumes that the existing maintenance terms for raw water (SPMRW3A(B)M), reactor building spray (IPMBSPA(B)M), decay heat removal pumps (LPM001A(B)M), and decay heat closed cycle cooling (SPMDHCA(B)M) are set to "FALSE" since the model logic does not preclude them from happening, even though the opposite train maintenance would not be performed during the AOT, and including the existing same train maintenance terms in the results would result in double counting.

Solving the model with the B train AOT maintenance event set to 1.0 results in an CDF of 2.61E-05/year.

For train B assumed out of service the result is:

$$\Delta CDF = CDF_{AOT} - CDF_{BASE} = 2.61E\text{-}05/yr - 4.99E\text{-}06/yr = \mathbf{2.11E\text{-}05/yr}$$

$$ICCDP = \Delta CDF * AOT = 2.11E\text{-}05/yr * 168\ hours * (1yr/8760\ hours) = \mathbf{4.05E\text{-}07}$$

The B train ICCDP case assumes a different plant alignment than that for the base case. The differences from the default MOR06 alignment are:

1. MUP-1A cooled by NSCCC
2. MUP-1C cooled by NSCCC
3. MUP-1B powered from ES-B
4. MUP ES Select=MUP1A/1B

This case assumes that the opposite train diesel (train A) is not out of service for maintenance since this is precluded by tech. specs. The analysis assumes that maintenance on High Pressure Injection (HPI), Emergency Feedwater (EFW), Auxiliary Feedwater (AFW), Emergency Feedwater Initiation and Control (EFIC), and Appendix R chiller (CHP-2) will not be performed concurrently with the extended AOT. It also assumes that the existing maintenance terms for raw water (SPMRW3A(B)M), reactor building spray (IPMBSPA(B)M), decay heat removal pumps (LPM001A(B)M), and decay heat closed cycle cooling (SPMDHCA(B)M) are set to "FALSE" since the model logic does not preclude them from happening, even though the opposite train maintenance would not be performed during the AOT, and including the existing same train maintenance terms in the results would result in double counting.

Per Reg Guide 1.177 changes in ICCDP less than 5E-7 are considered very small.

5.6 Incremental Conditional Large Early Release Probability

When the AOT is assumed to be occurring the instantaneous LERF value increases to 3.84E-07/yr for A train maintenance and 3.90E-7/yr for B train maintenance. Therefore:

Train A results:

$$\Delta LERF = LERF_{AOT} - LERF_{BASE} = 3.84E\text{-}07/yr - 3.69E\text{-}07/yr = \mathbf{1.5E\text{-}08/yr}$$

$$ICLERP = \Delta LERF * AOT = 1.5E\text{-}08/yr * 168\ hours * (1yr/8760\ hours) = \mathbf{2.88E\text{-}10}$$

Train B results:

$$\Delta LERF = LERF_{AOT} - LERF_{BASE} = 3.90E\text{-}07/yr - 3.69E\text{-}07/yr = \mathbf{2.1E\text{-}08/yr}$$

$$ICLERP = \Delta LERF * AOT = 2.1E\text{-}08/yr * 168\ hours * (1yr/8760\ hours) = \mathbf{4.03E\text{-}10}$$

Per Reg Guide 1.177 changes in ICLERP less than 5E-8 are considered very small.

5.7 External Events

The CR3 IPEEE (reference 3) and supporting data was reviewed to identify external event influences to the risk for the subject activities. The only potentially significant external events are fires and severe weather.

5.7.1 Fire Risk Sensitivity

A fire risk sensitivity study and a qualitative assessment for comparing shutdown verses online risk has been performed to help provide some insight.

CR3 does not have a fire PRA model that can be used to quantify the effect of the postulated fire scenario on LERF. However, because the predominant contributors to LERF for CR3 are scenarios based on steam generator tube ruptures (SGTR) or interfacing system LOCAs (ISLOCA), the LERF impact is estimated to be very low, because the low pressure injection and building spray systems are not significant mitigating systems for SGTR sequences. Therefore any increase in LERF due to fire will be very small.

Table 3 lists the fire zones identified as containing circuits applicable to the Decay Heat Removal (DH), DH Closed Cycle Cooling (DC), and Raw Water DC pumps and their supported front line systems. The Building Spray pumps are not included since they do not mitigate CDF. Table 3 displays the fire areas identified by the CR3 Appendix R fire study which are important. If the fire can be expected to impact both trains, or manual actions that are credited in the Fire Study, then the delta risk is expected to be minimal. The greatest risk impact due to train "A" being out of service is expected for fires which impact only the "B" train equipment, and similarly for the opposite train the greatest risk impact due to the "B" train being out of service is expected for fires which impact only the "A" train equipment. These zones are indicated with a "yes".

Table 4 and 5 provides the ignition sources and raw frequencies (reference 3) for each fire zone indicated as a candidate for PSA fire risk impact ("yes"). The compensated frequency column eliminates the contribution from transient sources in protected train rooms and equipment which will not be operated without special precautions. Motors that are not expected to be operating are also excluded from the compensated frequency. The excluded sources are shaded and italicized. If automatic suppression exists the IPEEE credit was applied. Finally a Conditional Core Damage Probability (CCDP) of 0.1 was applied as a sensitivity study for the purpose of estimating a core damage frequency. In other words, it was assumed that there was a 0.1 chance of core damage occurring given that a fire occurred. The instantaneous CDF due to fire based on these tables for "A" train out of service is 1.39E-04/yr and for "B" train is 2.49E-04/yr. This translates into an estimated CCDP associated with the 7 day AOT due to fire of:

"A"train 1.39E-04/yr * 7days * 1yr/365days = 2.67E-06

"B"train 2.49E-04/yr * 7days * 1yr/365days = 4.78E-06

These values could also be considered as a bounding delta CDF based on the fact that the assumed 0.1 CCDP is conservative. Additional compensatory actions such as dedicated fire watches could be used to further reduce this value.

A comparison of the on-line to off-line risk can not be directly quantified but can be addressed qualitatively. Although there is some level of increased risk performing this work on-line, the shutdown fire risk is expected to be greater due to transient initiating events. The risk due to fire is dominated by transient initiating events and during outages transient initiating event frequencies increase by an order of magnitude. The transient initiating event frequencies increase during an outage due to increased storage and maintenance. Furthermore, fire suppression by existing installed equipment and the fire brigade is impaired by maintenance activities that limit accessibility to the fire areas by staged equipment and scaffolding.

Table 3 – RW-DC Pump Related Fire Zones

ZONE	RWP-3A	DCP-1A	DHP-1A	RWP-3B	DCP-1B	DHP-1B	Fire Risk Impact due to Train A AOT	Fire Risk Impact due to Train B AOT
AB-75-4						x	Yes	Neg
AB-75-5			x				Neg	Yes
AB-95-3AA	x	xR	x				Neg	Yes
AB-95-3B	xW	xR	xW	xW	xR	xW	Minimal	Minimal
AB-95-3C			x				Neg	Yes
AB-95-3D				xW		xW	Yes	Neg
AB-95-3E	x		x				Neg	Yes
AB-95-3F	x		x				Neg	Yes
AB-95-3G	x	xR		xW	xR	xW	Neg (2)	Yes
AB-95-3K	x		x	xW		xW	Neg (2)	Yes
AB-95-3L		xR					Neg	Yes
AB-95-3M		xR					Neg	Yes
AB-95-3N		xR					Neg	Yes
AB-95-3P		xR					Neg	Yes
AB-95-3Q		xR					Neg	Yes
AB-95-3R		xR					Neg	Yes
AB-95-3T	x	xR					Neg	Yes
AB-95-3U	x	xR					Neg	Yes
AB-95-3W	x	xR	x				Neg	Yes
AB-95-3X				xW	xR	xW	Minimal (1)	Neg
AB-95-3Y			x			xR	Minimal (1)	Yes
AB-95-3Z	x	xR		xW	xR		Minimal	Yes
AB-119-6A		x			xW		Neg (2)	Yes
AB-119-6E		x					Neg	Yes
CC-95-101				x	x	x	Yes	Neg
CC-108-102	x	x	x	xW	xW	xW	Neg (2)	Yes
CC-108-103			xW	x	x	x	Yes	Neg
CC-108-104	x	x	x				Neg	Yes
CC-108-105	xW	xW	xW	x	x	x	Yes	Neg
CC-108-106	x	x	x	xM		xM	Minimal	Yes
CC-108-107				x	x	x	Yes	Neg
CC-108-108	x	x	x				Neg	Yes
CC-108-109	xM	xM	xM	xM	x	xM	Yes	Minimal
CC-108-110	x	x	x				Neg	Yes
CC-124-111	x		x	xM	xW	xM	Minimal (2)	Yes
CC-124-115				x	x	x	Yes	Neg
CC-124-116				x	x	x	Yes	Neg
CC-124-117	x	x	x				Neg	Yes
CC-134-118A	xT	xT	xT	xT	xT	xT	Minimal	Minimal
CC-145-118B	xT	xT	xT	xT	xT	xT	Minimal	Minimal
CC-164-121	xT	xT	xT	xT	xT	xT	Minimal	Minimal

x -indicates equipment not available due to fire
xW -indicates protected with fire wrap
xM -indicates available with manual actions
xT -indicates available from remote shutdown panel
xR -Appendix R credits equipment repair for long term availability
(1) -Classified as minimal based on Appendix R (hardware repair).
(2) -Credit given for fire wrap to reduce significance.

Table 4 – Ignition Sources/Frequencies for Impacted Areas for Train A

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
AB-75-4	*TRANS-4*	9.73E-05					
	BSP-1B	5.21E-05					
	DHP-1B	5.21E-05	2.02E-04	0.00E+00	1	0.1	0.00E+00
AB-95-3D	TRANS-3D	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
CC-95-101A	TRANS-101A	9.73E-05	9.73E-05	9.73E-05	0.05	0.1	4.87E-07
CC-95-101B	TRANS-101B	9.73E-05	9.73E-05	9.73E-05	0.05	0.1	4.87E-07
CC-95-101C	TRANS-101C	9.73E-05	9.73E-05	9.73E-05	0.05	0.1	4.87E-07
CC-108-103	TRANS-103	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
CC-108-105	AHF-71	1.85E-05					
	TRANS-105B	4.86E-05					
	DPBC-1F	8.89E-05					
	ACTR-14	1.20E-05					
	TRANS-105-A	4.86E-05					
	DPBC-1D	8.89E-05					
	DPBC-1B	8.89E-05					
	AHDP-12	4.48E-06					
	DPDP-1B	4.48E-06	4.03E-04	4.03E-04	1	0.1	4.03E-05
CC-108-107	MTSW-2F R3	7.20E-06					
	MTSW-2E R2	7.20E-06					
	MTSW-2F R2	7.20E-06					
	MTSW-2F R1	7.20E-06					
	MTSW-2E R3	7.20E-06					
	MTSW-2E R7	7.20E-06					
	MTSW-2F R4	7.20E-06					
	MTSW-2E R6	7.20E-06					
	RCMP-3B	7.20E-06					
	MTSW-2E R5	7.20E-06					
	MTSW-2F R5	7.20E-06					
	MTSW-2F R7	7.20E-06					
	MTSW-2F R6	7.20E-06					
	TRANS-107-C	1.95E-05					
	RSD AUX B R	7.20E-06					
	TRANS-107-E	1.95E-05					

Table 4 – Ignition Sources/Frequencies for Impacted Areas for Train A

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	TRANS-107-D	1.95E-05					
	TRANS-107-B	1.95E-05					
	RSD RLY B1	7.20E-06					
	TRANS-107-A	1.95E-05					
	RSD RLY B	7.20E-06					
	MTSW-2E R1	7.20E-06					
	MTSW-2E R1	7.20E-06	2.27E-04	2.27E-04	1	0.1	2.27E-05
CC-108-109	VBIT-3E	4.48E-06					
	VBTR-3B	1.20E-05					
	VBTR-3D	1.20E-05					
	VBXS-1E	4.48E-06					
	VBTR-3E	1.20E-05					
	VBIT-1D	4.48E-06					
	TRANS-109-A	4.86E-05					
	TRANS-109-B	4.86E-05					
	VBXS-1B	4.48E-06					
	VBXS-1D	4.48E-06					
	VBTR-2D	1.20E-05					
	VBDP-14	4.48E-06					
	VBXS-3D	4.48E-06					
	VBTR-2B	1.20E-05					
	VBXS-3B	4.48E-06					
	VBDP-15	4.48E-06					
	VBTR-2E	1.20E-05					
	VBIT-1B	4.48E-06	2.14E-04	2.14E-04	1	0.1	2.14E-05
CC-124-115	VBTR-4B	4.48E-06					
	EFIC CAB B	4.48E-06					
	TRANS-115	9.73E-05					
	AHF-54B	1.85E-05					
	VBDP-10	1.85E-05					
	RR5B2	4.48E-06					
	RR4B1	4.48E-06					
	RR4B	4.48E-06					
	EFIC CH B	4.48E-06	1.47E-04	1.47E-04	1	0.1	1.47E-05
CC-124-116	TRANS-116-H	1.22E-05					

Table 4 – Ignition Sources/Frequencies for Impacted Areas for Train A

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	AHF-76	1.85E-05					
	AHF-77	1.85E-05					
	MTSW-3G	1.20E-05					
	MTSW-3G R1	7.20E-06					
	TRANS-116-A	1.22E-05					
	TRANS-116-B	1.22E-05					
	TRANS-116-C	1.22E-05					
	TRANS-116-D	1.22E-05					
	TRANS-116-E	1.22E-05					
	TRANS-116-F	1.22E-05					
	TRANS-116-G	1.22E-05					
	MTSW-3G R2	7.20E-06					
	MTSW-3G R3	7.20E-06					
	DPDP-5B	7.20E-06					
	DPDP-8D	7.20E-06					
	RC RCITS-B	7.20E-06	1.90E-04	1.90E-04	1	0.1	1.90E-05
						Total	1.39E-04

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
AB-75-5	*TRANS-5*	9.73E-05					
	BSP-1A	5.21E-05					
	DHP-1A	5.21E-05	2.02E-04	0.00E+00	1	0.1	0.00E+00
AB-95-3AA	TRANS-3AA	9.73E-05					
	MUP-1B	5.21E-05	1.49E-04	1.49E-04	0.02	0.1	2.98E-07
AB-95-3C	MCC MUV-23/24	4.48E-06					
	MCC MUV-25/26	4.48E-06					
	TRANS-3C	9.73E-05	1.06E-04	1.06E-06	0.02	0.1	2.12E-07
AB-95-3E	*MUP-1A*	5.21E-05					
	TRANS-3E	9.73E-05	1.49E-04	9.73E-05	0.02	0.1	1.95E-07
AB-95-3F	*MUP-1C*	5.21E-05					
	TRANS-3F	9.73E-05	1.49E-04	9.73E-05	1	0.1	9.73E-06
AB-95-3G	MTMC-3 R6	4.48E-06					
	WDTR-1	1.20E-05					
	RMA-3	4.48E-06					
	MTMC-6 R6	4.48E-06					
	MTMC-6 R5	4.48E-06					
	MTMC-3 R10	4.48E-06					
	MTMC-3 R12	4.48E-06					
	MTMC-3 R13	4.48E-06					
	MTMC-3 R14	4.48E-06					
	MTMC-3 R2	4.48E-06					
	MTMC-3 R3	4.48E-06					
	MTMC-3 R4	4.48E-06					
	MTMC-6 R4	4.48E-06					
	MTMC-3 R7	4.48E-06					
	MTMC-3 R8	4.48E-06					
	MTMC-6 R3	4.48E-06					
	MTMC-3 R9	4.48E-06					
	MTMC-6 R2	4.48E-06					
	MTMC-6 R1	4.48E-06					
	WDCP-2	4.48E-06					
	WDCP-1	4.48E-06					
	WASTE DISP	4.48E-06					
	WASTE DISP	4.48E-06					
	MTMC-3 R1	4.48E-06					
	MTMC-6 R5	4.48E-06					

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	MTMC-18 R2	4.48E-06					
	MTMC-18 R6	4.48E-06					
	MTMC-18 R5	4.48E-06					
	MTMC-18 R4	4.48E-06					
	MTMC-18 R3	4.48E-06					
	HTTR-4B	1.20E-05					
	MTMC-18 R7	4.48E-06					
	MTMC-18 R8	4.48E-06					
	MTMC-18 R9	4.48E-06					
	MTMC-19 R1	4.48E-06					
	MTMC-19 R2	4.48E-06					
	TRANS-3G-B	4.86E-05					
	MTMC-19 R4	4.48E-06					
	MTMC-19 R5	4.48E-06					
	TRANS-3G-A	4.86E-05					
	HTCP-4	4.48E-06					
	HTCP-1	4.48E-06					
	MTMC-19 R3	4.48E-06					
	MTMC-3 R11	4.48E-06					
	MTMC-18 R10	4.48E-06					
	MTMC-19 R9	4.48E-06					
	MTMC-19 R8	4.48E-06					
	MTMC-19 R7	4.48E-06					
	MTMC-19 R6	4.48E-06					
	MTMC-18 R1	4.48E-06	3.27E-04	3.27E-04	0.02	0.1	6.54E-07
AB-95-3K	TRANS-3K	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
AB-95-3L	TRANS-3L	9.73E-05					
	ASP-2B	5.21E-05					
	ASP-2A	5.21E-05	2.02E-04	9.73E-05	1	0.1	9.73E-06
AB-95-3M	TRANS-3M	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
AB-95-3N	TRANS-3N	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
AB-95-3P	*WDP-12A*	5.21E-05					
	WDP-12B	5.21E-05					
	WDP-13A	5.21E-05					
	WDP-13B	5.21E-05					
	TRANS-3P	9.73E-05	3.06E-04	9.73E-05	1	0.1	9.73E-06
AB-95-3Q	TRANS-3Q	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
AB-95-3R	TRANS-3R	9.73E-05					

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	HTTR-4A	1.20E-05					
	HAYES CAB	4.48E-06					
	HTDP-1B	4.48E-06					
	HTDP-4A	4.48E-06					
	HTTR-1A	1.20E-05					
	HTTR-1B	1.20E-05					
	HTDP-1A	4.48E-06					
	WDP-1B	6.71E-05	2.18E-04	2.18E-04	1	0.1	2.18E-05
AB-95-3T	TRANS-3T	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
AB-95-3U	TRANS-3U	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
AB-95-3W	*WDP-5C*	5.21E-05					
	WDP-5B	5.21E-05					
	WDP-5A	5.21E-05					
	TRANS-3W	9.73E-05	2.54E-04	9.73E-05	1	0.1	9.73E-06
AB-119-6A	HY-6A	8.00E-04					
	TRANS-6A-H	1.22E-05					
	TRANS-6A-G	1.22E-05					
	TRANS-6A-F	1.22E-05					
	TRANS-6A-E	1.22E-05					
	TRANS-6A-D	1.22E-05					
	TRANS-6A-B	1.22E-05					
	TRANS-6A-A	1.22E-05					
	TRANS-6A-C	1.22E-05	8.98E-04	8.98E-04	0.02	0.1	1.80E-06
AB-119-6E	MTMC-4 R9	4.48E-06					
	MTMC-4 R10	4.48E-06					
	MTMC-4 R11	4.48E-06					
	MTMC-4 R2	4.48E-06					
	MTMC-4 R4	4.48E-06					
	MTMC-4 R6	4.48E-06					
	MTMC-4 R8	4.48E-06					
	MTMC-4 R1	4.48E-06					
	TRANS-6E	9.73E-05					
	MTMC-4 R3	4.48E-06					
	MTMC-4 R7	4.48E-06					
	HY-6E	8.00E-04					
	MTMC-4 R5	4.48E-06					
	MTMC-21 R1	4.48E-06					
	MTMC-21 R2	4.48E-06					
	MTMC-21 R3	4.48E-06					

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	MTMC-21 R6	4.48E-06					
	MTMC-21 R4	4.48E-06					
	MTMC-21 R5	4.48E-06	9.73E-04	9.73E-04	0.02	0.1	1.95E-06
CC-108-102	TRANS-102	9.73E-05					
	AHF-69	1.85E-05					
	REMOTE SHUTDOWN PNL	4.48E-06	1.20E-04	1.20E-04	1	0.1	1.20E-05
CC-108-104	TRANS-104	9.73E-05	9.73E-05	9.73E-05	1	0.1	9.73E-06
CC-108-106	TRANS-106-A	9.73E-05					
	DPBC-1A	8.89E-05					
	DPBC-1E	8.89E-05					
	DPDP-1A	4.48E-06					
	DPBC-1C	8.89E-05	3.68E-04	3.68E-04	1	0.1	3.68E-05
CC-108-108	MTSW-2C R4	7.20E-06					
	AHF-72	1.85E-05					
	MTTR-4	7.20E-06					
	MTSW-2D R6	7.20E-06					
	MTSW-2D R7	7.20E-06					
	MTSW-2D R5	7.20E-06					
	CAIT-1	7.20E-06					
	MTSW-2D R3	7.20E-06					
	MTSW-2D R2	7.20E-06					
	MTSW-2D R1	7.20E-06					
	MTSW-2C R6	7.20E-06					
	MTSW-2C R3	7.20E-06					
	MTSW-2C R2	7.20E-06					
	MTSW-2C R1	7.20E-06					
	DPDP-8A	7.20E-06					
	MTSW-2C R7	7.20E-06					
	RSD RLY A	7.20E-06					
	MTSW-2D R4	7.20E-06					
	RSD AUX A RLY	7.20E-06					
	RCMP-3A	7.20E-06					
	RSD RLY A1	7.20E-06					
	TRANS-108-B	4.86E-05					
	TRANS-108-	4.86E-05	2.60E-04	2.60E-04	1	0.1	2.60E-05

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	A						
CC-108-110	VBIT-1A	4.48E-06					
	VBIT-1C	4.48E-06					
	VBXS-3A	4.48E-06					
	VBXS-1A	4.48E-06					
	VBXS-1C	4.48E-06					
	VBTR-3C	1.20E-05					
	VBTR-3A	1.20E-05					
	VBXS-3C	4.48E-06					
	VBDP-12	4.48E-06					
	TRANS-110	9.73E-05					
	VBTR-2C	1.20E-05					
	AHHE-55	4.48E-06					
	AHHE-54	4.48E-06					
	VBDP-13	4.48E-06					
	VBTR-2A	1.20E-05	1.90E-04	1.90E-04	1	0.1	1.90E-05
CC-124-111	DRRD7-6A	4.48E-06					
	DRRD3-8	4.48E-06					
	DRRD7-5A	4.48E-06					
	DRRD6B	4.48E-06					
	DRRD7-7A	4.48E-06					
	DRRD7-6B	4.48E-06					
	DRRD6A	4.48E-06					
	DRRD7-5B	4.48E-06					
	DRRD4-1	4.48E-06					
	DRRD2-3	4.48E-06					
	DRRD3-1	4.48E-06					
	DRRD3-2	4.48E-06					
	DRRD3-3	4.48E-06					
	DRRD3-4	4.48E-06					
	DRRD3-5	4.48E-06					
	DRRD3-6	4.48E-06					
	DRRD4-2	4.48E-06					
	DRTR-1B	1.20E-05					
	DRRD5R	4.48E-06					
	DRRD7-7B	4.48E-06					
	DRRD4-3	4.48E-06					
	DRRD4-4	4.48E-06					

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	DRRD4-5	4.48E-06					
	DRRD4-7	4.48E-06					
	LIGHTING XFMR A	1.20E-05					
	DRRD5L	4.48E-06					
	DRRD3-7	4.48E-06					
	TRANS-111-L	6.49E-06					
	DRRD8B	4.48E-06					
	TRANS-111-F	6.49E-06					
	TRANS-111-G	6.49E-06					
	TRANS-111-H	6.49E-06					
	TRANS-111-I	6.49E-06					
	TRANS-111-D	6.49E-06					
	TRANS-111-K	6.49E-06					
	TRANS-111-C	6.49E-06					
	TRANS-111-M	6.49E-06					
	TRANS-111-N	6.49E-06					
	TRANS-111-O	6.49E-06					
	LIGHTING XFMR B	1.20E-05					
	MUX 4	4.48E-06					
	TRANSMITTER PWR SUPP CAB A,AB, B	4.48E-06					
	TRANS-111-J	6.49E-06					
	DRTR-1A	1.20E-05					
	DRRD7-8B	4.48E-06					
	DRRD8A	4.48E-06					
	VBTR-1A	1.20E-05					
	VBTR-1B	1.20E-05					
	RRHV	4.48E-06					
	TRANS-111-E	6.49E-06					
	DRRD2-2	4.48E-06					
	DRRD7-8A	4.48E-06					
	DRRD4-6	4.48E-06					
	EHCC-1	4.48E-06					

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	EHCC-2	4.48E-06					
	EHCC-3	4.48E-06					
	TRANS-111-A	6.49E-06					
	TRANS-111-B	6.49E-06					
	RRPSA	4.48E-06					
	PAX CAB	4.48E-06					
	MUX 2	4.48E-06					
	CDR VOLTAGE REG B	4.48E-06					
	CRDM GROUP POWER SUPPLY CAB	4.48E-06					
	MUX 1	4.48E-06					
	MUX 5	4.48E-06					
	DRRD4-8	4.48E-06					
	ACTR (NEAR JAIL DOOR)	1.20E-05					
	DRRD2-1	4.48E-06					
	CDR VOLTAGE REG A	4.48E-06					
	COMM CAB (PAX)	4.48E-06					
	DPDP-4B	4.48E-06					
	DPDP-4A	4.48E-06					
	COMTEL 2020 REMOTE	4.48E-06					
	CRD BKR A CAB	4.48E-06					
	CRD BKR B CAB	4.48E-06					
	ACTR (SW CORNER)	1.20E-05					
	RR3A	4.48E-06					
	AHF-54A	1.85E-05					
	AHDP-11	4.48E-06					
	RR3B	4.48E-06					
	RR3	4.48E-06					
	ACTR-15	1.20E-05					
	RR2AB	4.48E-06					

Table 5 – Ignition Sources/Frequencies for Impacted Areas for Train B

ZONE	SOURCES IPEEE	IGNF SOURCE	IGNF ZONE	IGNF ZONE COMP	Suppression Credit	CCDP due to RWP-3B	CDF
	RR1B	4.48E-06					
	RR1	4.48E-06					
	RFM	4.48E-06					
	MUX 3	4.48E-06					
	RR2	4.48E-06	5.06E-04	5.06E-04	0.02	0.1	1.01E-06
CC-124-117	TRANS-117-B	1.39E-05					
	MTSW-3F R1	7.20E-06					
	MTSW-3F R3	7.20E-06					
	RC RCITS-C	7.20E-06					
	TRANS-117-G	1.39E-05					
	MTSW-3F R2	7.20E-06					
	DPDP-5A	7.20E-06					
	MTSW-3F	1.20E-05					
	TRANS-117-A	1.39E-05					
	AHF-75	1.85E-05					
	AHF-74	1.85E-05					
	ES MCC 3AB/TS	7.20E-06					
	TRANS-117-C	1.39E-05					
	TRANS-117-D	1.39E-05					
	RC RCITS-A	7.20E-06					
	DPDP-8C	7.20E-06					
	TRANS-117-E	1.39E-05					
	TRANS-117-F	1.39E-05	2.04E-04	2.04E-04	1	0.1	2.04E-05
						Total	2.49E-04

5.7.2 Weather Sensitivity

The chances of severe weather are greater in Florida during the summer months, and the main impact of severe weather is an increased probability for loss of offsite power. A sensitivity case was performed and demonstrated a minimal increase in risk due to a higher loss of offsite power frequency during the extended AOT.

To evaluate the sensitivity to weather events the frequency of losing offsite power was increased to assess the impact of severe weather. This involved increasing both the normal LOOP and partial LOOP initiating events (T3, T15) in the PRA model by a factor of 3. The increase was not significant.

Results:

A new base CDF was quantified by increasing the normal and partial Loss of Offsite Power (LOOP) initiating events (T3, T15) by a factor of three resulting in a CDF of 5.46E-06/year

Solving the model with the A train AOT maintenance event set to 1.0 and the normal and partial LOOP initiating events (T3, T15) increased by a factor of three resulted in an CDF of 2.20E-05/year.

Solving the model with the B train AOT maintenance event set to 1.0 and the normal and partial LOOP initiating events (T3, T15) increased by a factor of three resulted in a CDF of 2.85E-05/year.

ICCDP is equal to the integration of the change in risk over time. For this assessment it can be determined that the increase in risk when the AOT is occurring multiplied by the duration of the AOT. For train A assumed out of service the result is:

$$\Delta CDF = CDF_{AOT} - CDF_{BASE} = 2.20E\text{-}05/yr - 5.46E\text{-}06/yr = 1.65E\text{-}05/yr$$

$$ICCDP_{weather} = \Delta CDF * AOT = 1.65E\text{-}05/yr * 168 \text{ hours} * (1yr/8760 \text{ hours}) = \mathbf{3.16E\text{-}07}$$

From Section 5.5, the ICCDP for train "A" assumed out of service was 2.97E-7 so the change in ICCDP is:

$$\Delta ICCDP = ICCDP_{weather} - ICCDP_{base} = 3.16E\text{-}07 - 2.97E\text{-}7 = 1.90E\text{-}08$$

For train B assumed out of service the result is:

$$\Delta CDF = CDF_{AOT} - CDF_{BASE} = 2.85E\text{-}05/yr - 5.46E\text{-}06/yr = 2.30E\text{-}05/yr$$

$$ICCDP_{weather} = \Delta CDF * AOT = 2.30E\text{-}05/yr * 168 \text{ hours} * (1yr/8760 \text{ hours}) = \mathbf{4.41E\text{-}07}$$

From Section 5.5, the ICCDP for train "B" assumed out of service was 4.05E-7 so the change in ICCDP is:

$$\Delta ICCDP = ICCDP_{weather} - ICCDP_{base} = 4.41E\text{-}07 - 4.05E\text{-}7 = 3.60E\text{-}08$$

The ΔICCDP due to the increase in of the LOOP initiating events was less than 1E-7 which shows the results are not sensitive to external events that increase the frequency of LOOP.

5.8 Intake Bay Maintenance Sensitivity

This sensitivity will determine if intake bay work should be avoided during the 7 day AOT. Intake bay work would remove additional pumps from service. This sensitivity will assume the bay work will take 3 days, and compare the total ICCDP of doing a seven day AOT and a separate intake bay outage versus overlapping the intake bay work with the seven day AOT.

"A" Train

Solving the model with the A train AOT maintenance event set to 1.0 results in a CDF of 2.05E-05/year from Section 5.5. Subtracting the base CDF from the A train AOT CDF results in a ΔCDF_{AOT} of 1.55E-5/year.

Solving the model with the A train Intake Bay maintenance which affects RWP-2A and RWP-3A set to 1.0 results in a CDF of 2.59E-05/year. Subtracting the base CDF from the A train Intake Bay maintenance CDF results in a ΔCDF_{intake} of 2.09E-5/year.

Solving the model with the A train Intake Bay maintenance and AOT maintenance set to 1.0 results in a CDF of 2.59E-05/year. Subtracting the base CDF from the A train Intake Bay maintenance and AOT CDF results in a $\Delta CDF_{intake+AOT}$ of 2.09E-5/year.

"A" Train Separate AOT and Intake Bay Evaluation

$ICCDP_{sep} = (\Delta CDF_{intake} * 3\ days) + (\Delta CDF_{AOT} * 7\ days)$

$ICCDP_{sep} = (2.09E\text{-}05 * 3\ days * [1yr/365\ days]) + (1.55E\text{-}05/year * 7Days * [1yr/365])$
$ICCDP_{sep} = \mathbf{4.69E\text{-}07}$

"A" Train Concurrent AOT and Intake Bay Evaluation

$ICCDP_{con} = (\Delta CDF_{intake+AOT} * 3\ days) + (\Delta CDF_{AOT} * 4\ days)$

$ICCDP_{con} = (2.09E\text{-}5 * 3\ days * [1yr/365\ days]) + (1.55E\text{-}05/year * 4\ days * [1yr/365])$
$ICCDP_{con} = \mathbf{3.42E\text{-}7}$

"B" Train

The "B" train runs are performed using a truncation limit of 1E-10, because computer limitation at a 1E-12 truncation. This results in a new base CDF of 4.16E-6 without AOT maintenance.

Solving the model with the B train AOT maintenance event set to 1.0 results in a CDF of 2.49E-05/year similar to Section 5.5, but with a 1E-10 truncation limit. Subtracting the base CDF from the B train AOT CDF results in a ΔCDF_{AOT} of 2.07E-5/year.

Solving the model with the B train Intake Bay maintenance which affects RWP-1, RWP-2B, and RWP-3B results in a CDF of 5.68E-05/year. Subtracting the base CDF from the B train Intake Bay maintenance CDF results in a ΔCDF_{intake} of 5.26E-5/year.

Solving the model with the B train Intake Bay maintenance and AOT maintenance results in a CDF of 5.68E-05/year. Subtracting the base CDF from the B train Intake Bay maintenance and AOT CDF results in a $\Delta CDF_{intake+AOT}$ of 5.26E-5/year.

"B" Train Separate AOT and Intake Bay Evaluation

$ICCDP_{sep} = (\Delta CDF_{intake} * 3\ days) + (\Delta CDF_{AOT} * 7\ days)$

$ICCDP_{sep} = (5.26E\text{-}5 * 3\ days * [1yr/365\ days]) + (2.07E\text{-}05/year * 7Days * [1yr/365])$
$ICCDP_{sep} = \mathbf{8.29E\text{-}7}$

"B" Train Concurrent AOT and Intake Bay Evaluation

$ICCDP_{con} = (\Delta CDF_{intake} * 3\ days) + (\Delta CDF_{AOT} * 4\ days)$

$ICCDP_{con} = (5.26E\text{-}5 * 3\ days * [1yr/365\ days]) + (2.07E\text{-}05/year * 4\ days * [1yr/365])$
$ICCDP_{con} = \mathbf{6.59E\text{-}7}$

The "B" train ICCDP case assumes a different plant alignment than that for the base case. The differences from the default MOR06 alignment are:

1. MUP-1A cooled by NSCCC
2. MUP-1C cooled by NSCCC
3. MUP-1B powered from ES-B
4. MUP ES Select=MUP1A/1B

Results of Intake Bay Evaluation

This sensitivity shows that if maintenance is required on the intake bays and a decay heat outage is also planned, doing these activities at the same time would reduce the total risk to the public. These results show that intake bay B maintenance should be minimized and carefully evaluated, regardless of whether the proposed 7 day AOT technical specification change is implemented or not.

6.0 Results / Conclusions

The results of the cutsets for the CDF, LERF, and various sensitivity cases were each reviewed in order to assure that the resulting accident sequences gave reasonable results for the various configurations examined for this study.

The ΔCDF of 5.5E-07/yr is below the Reg. Guide 1.174 threshold of 1E-06 and is considered very small risk. Also the ΔLERF of 1E-09/yr is below the Reg. Guide threshold of 1E-07 and is considered very small risk.

Comparing the ICCDP criteria from Reg. Guide 1.177 for the "A" and "B" trains resulted in a value of 2.97E-7 for the "A" train and 4.05E-7 for the "B" train. The ICCDP criteria was not exceeded for either train. The ICLERP criteria were also not exceeded for either train AOT.

Doing maintenance on the decay heat removal system while at power, rather than during shutdown when it is the primary source of cooling, the risk during shutdown is reduced. Also, as discussed in the topical report [ref. 5] preventative maintenance can be scheduled more frequently, enhancing the overall reliability of equipment and reducing the number of entries into LCOs. A longer AOT also allows more flexibility in work scheduling which leads to more orderly completion of maintenance.

Compensatory measures will also be implemented in order to reduce the risk impact of the proposed AOT extension. These measures include:

1. Avoid simultaneous outages of additional risk-significant equipment during the proposed AOT extension. The components whose simultaneous unavailability are to be avoided , in addition to the current TS requirements, are EFW, AFW, EFIC, HPI, Appendix R Cooler (CHP-2), and their power supplies.
2. Makeup pump power, cooling, and ES alignment can affect the risk results, alignment configuration should be considered when planning the proposed AOT extension.
3. Minimize concurrent maintenance on RWP-2A(B) with the proposed AOT extension.
4. Defining specific criteria for scheduling only those preventative maintenance procedures. which can be completed within the proposed AOT extension, such that the chance for a forced outage due to failure to complete the maintenance is negligible.
5. Assuring that the frequency of entry into the proposed AOT extension, and consequently the average maintenance duration per year, remains within that assumed in this analysis.
6. Establish a periodic fire watch and limit transient combustibles in the opposite train decay heat pump vault.
7. This assessment is based on appropriate use of risk management actions including limiting work on defense-in-depth equipment and other risk significant systems. The results of this calculation should not be used in lieu of performing a pre-maintenance risk assessment. Risk assessment should be performed prior to and at the time the proposed AOT extension is entered.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time

ATTACHMENT F

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Florida Power Corporation (FPC) in this document. Any other actions discussed in the submittal represent intended or planned actions by FPC. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

Commitment	Due Date
CR-3 will perform procedure CP-253, "Power Operation Risk Assessment and Management" which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on core damage frequency. CR-3 will not plan any maintenance that results in "Higher Risk" (Orange Color Code) during an extended outage (greater than 72 hours) of the LPI, BS, DC or RW System.	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW System
The opposite train of EFW, Auxiliary Feedwater System, Emergency Feedwater Initiation and Control System, HPI, Appendix R Cooler, and their power supplies will be administratively designated as "protected" (i.e., no planned maintenance or discretionary equipment manipulation).	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW System
CR-3 will not initiate an extended preventive maintenance outage (greater than 72 hours) on the LPI, BS, DC or RW System if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW System
When extended maintenance (greater than 72 hours) is scheduled on a train of the LPI or BS System, CR-3 will limit transient combustibles in the decay heat pump vault of the opposite train and establish a periodic fire watch of the decay heat pump vault of the opposite train.	During extended (greater than 72 hours) preplanned outage on the LPI or BS System
When extended maintenance (greater than 72 hours) is scheduled on a train of the DC or RW System, CR-3 will limit transient combustibles in the seawater room and establish a periodic fire watch in the seawater room.	During extended (greater than 72 hours) preplanned outage on the DC or RW System