April 30, 2008

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING EXTENSION OF ALLOWED OUTAGE TIME TO 7 DAYS AND ELIMINATION OF SECOND COMPLETION TIMES (TAC NO. MD5241)

Dear Mr. Young:

The Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 229 to Facility Operating License No. DPR-72 for Crystal River Unit 3 in response to your letter dated April 13, 2007, and as supplemented by letters dated September 4 and 13, 2007, and February 25, 2008. The amendment changes the technical specifications (TSs) to extend the completion time associated with an inoperable low-pressure injection train, reactor building spray train, decay heat closed cycle cooling water train, and decay heat seawater train, from 72 hours to 7 days. The change has been requested consistent with NRC-approved TS Task Force (TSTF) traveler TSTF-430, Revision 2. Additional TS changes implement TSTF-439, Revision 2, to eliminate second completion times. In addition to implementing the above technical changes, editorial and administrative changes to the TSs eliminate footnotes associated with one-time only changes and add the word “required” to describe “trains” in TS 3.6.6.

A copy of the safety evaluation is enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 229 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

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CITY OF NEW SMYRNA BEACH

CITY OF OCALA

ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO

SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 229
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated April 13, 2007, as supplemented by letters dated September 4 and 13, 2007, and February 25, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Facility Operating License (FOL) and Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C. (2) of FOL No. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 229 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: April 30, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 229

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of Facility Operating License DPR-72 with the attached revised pages.

Remove	Insert
4	4
5a	5a
5b	5b

Replace the following pages of the Appendix "A" Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
1.3-2	1.3-2
1.3-6	1.3-6
1.3-7	1.3-7
3.5-4	3.5-4
3.6-17	3.6-17
3.6-18	3.6-18
3.7-9	3.7-9
3.7-17	3.7-17
3.7-21	3.7-21
3.8-2	3.8-2
3.8-3	3.8-3
3.8-31	3.8-31
3.8-32	3.8-32

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated April 13, 2007 (Ref. 1), and as supplemented by letters dated September 4, 2007 and September 13, 2007 (Ref. 3), and February 25, 2008 (Ref. 4) Florida Power Corporation (FPC, the licensee), submitted a license amendment request (LAR) regarding the Crystal River Unit 3 (CR-3) Technical Specifications (TSs) and Facility Operating License (FOL).

The supplements dated September 4 and 13, 2007, and February 25, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC, Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* September 12, 2007 (72 FR 52167).

1.1 Proposed License Amendment

The licensee is proposing to implement the following NRC-approved Technical Specification Task Force (TSTF) travelers:

- TSTF-430, Revision 2, "AOT [Allowed Outage Time] Extension to 7 Days for LPI [Low-Pressure Injection] and Containment Spray (BAW-2995-A, Rev. 1)" (Ref. 5).
- TSTF-439, Revision 2, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO [Limiting Condition for Operation]" (Ref. 6).
- Editorial/administrative changes

The proposed changes associated with TSTF-430 would increase the completion times (CTs) associated with one inoperable LPI train (TS 3.5.2); one Reactor Building Spray (RBS) train (TS 3.6.6); one Decay Heat Closed Cycle Cooling Water (DC) train (TS 3.7.8); and one Decay Heat Seawater (RW, or RW-DC) train (TS 3.7.10); from 72 hours to 7 days.

The proposed changes associated with TSTF-439 would eliminate the second CT for the RBS (TS 3.6.6), Emergency Feedwater (EFW) (TS 3.7.5), AC Sources - Operating (TS 3.8.1), and Distribution Systems - Operating (TS 3.8.9).

In addition to implementing the above technical changes, editorial and administrative changes to TSs are also proposed, which eliminate footnotes associated with one-time only changes and add the word "required" to describe "trains" in TS 3.6.6.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations for Deterministic Evaluations

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. These TSs are derived from the plant safety analyses.

The Commission's regulatory requirements related to the contents of TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36 which ensures that TSs specified LCOs consistent with assumed values of the initial conditions in the licensee's safety analyses. In accordance with the 10 CFR 50.36, the staff and the Nuclear Steam Supply System Owner's Groups developed improved standard TSs (ISTSs), which meet 10 CFR 50.36 (d)(2)(ii) and 10 CFR 50.36 (d)(3)(ii) requirements. The licensee is using the guidance from the NRC-approved NUREG-1430, Revision 3, "Standard Technical Specifications (STSs) Babcock and Wilcox Plants," (Ref. 7), and the guidance from NUREG-800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," (Ref.8), as appropriate for their plant.

In general, there are two classes of changes to TSs: (1) changes needed to reflect contents of the design basis (TSs are derived from the design basis), and (2) voluntary changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs over time. This amendment deals with the second class of change; namely, changes that reflect the evolution in policy and guidance as to the preferred format for TSs.

Licensees may revise the TSs to adopt ISTSs format and content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards.

In NUREG 1430, a second CT was included for certain 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 LCO. These CTs (henceforth referred to as "second Completion Times") are joined by an "AND" logical connector to the condition-specific CT and state "X days from discovery of failure to meet the LCO" (where "X" varies by specification). The intent of the second CT was to preclude entry into and out of the ACTIONS for an indefinite period of time without meeting the LCO by providing a limit on the amount of time that the LCO could not be met for various combinations of conditions. TSTF-439, Revision 2 deletes these second CTs from the affected required actions from the STSs.

On June 20, 2005, the commercial nuclear electrical power generation industry owners group TSTF submitted a proposed change, TSTF-439, Revision 2, to the ISTSs on behalf of the industry. TSTF-439, Revision 2 was approved by the NRC in a letter to the TSTF, dated January 11, 2006 (Ref. 9)

The Babcock and Wilcox (BAW, B&W) Topical Report BAW-2295A, Revision 1 (Ref. 10), results showed that the risk significance from extending the proposed completion time for an inoperable LPI train or an inoperable RBS system from 72 hours to 7 days was small and within the Regulatory Guides (RG)1.174 (Ref. 11) and 1.177 (Ref. 12) guidance.

Risk-informed improvements to TSs are intended to maintain or improve safety while reducing unnecessary burden, and to bring TSs 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).

2.1.1 Adjusting Completion Times and Surveillance Intervals

Guidance documents have been prepared to assist in requesting risk-informed allowed outage time (also called completion time) and surveillance test interval extensions. These efforts (categorized as "Option 1" in the framework of the Risk-Informed Regulatory Improvement Program) have resulted in risk-informed amendments at numerous plants, and owners groups continue to submit topical reports to support additional applications.

2.1.2 Risk Management Technical Specifications

Issuance of the Maintenance Rule, 10 CFR 50.65, in July 1991 marked the advent of a regulation with significant implications for the evolution of TSs. Prior to 10 CFR 50.65, TSs were the primary rules governing operations, including what equipment must normally be in service, how long equipment can be out-of-service, compensatory actions, and surveillance testing to demonstrate equipment readiness. The goal of these TSs is to provide adequate assurance of the availability and reliability of equipment needed to prevent, and if necessary mitigate, accidents and transients. The Maintenance Rule shares this same goal but operates at a more fundamental level with a dynamic and more comprehensive process.

2.1.3 The Maintenance Rule

The Maintenance Rule requires each licensee to monitor the performance or condition of structures, systems, and components (SSCs) against licensee-established goals to ensure that the SSCs are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety, and where practical, take into account industry-wide operating experience. If the performance or condition of an SSC does not meet established goals, appropriate corrective action is required to be taken. The effectiveness of these performance monitoring activities, and associated corrective actions, is evaluated at least every refueling cycle, not to exceed 24 months.

2.2 Applicable Risk Informed Regulatory Criteria/Guidelines

The RGs on which the NRC staff based its acceptance are:

- RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.
- RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in AOTs. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensees evaluation of the risk associated with a proposed CT TS change, as discussed below.
    - Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission’s Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses probabilistic risk assessment (PRA) quality, including the technical adequacy of the licensee’s plant-specific PRA for the subject application. Cumulative risk of the proposed TS change in light of past related applications or additional applications under review is also considered along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.
    - Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.
    - Tier 3 addresses the licensee’s overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk-significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance,

subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the SRP (Ref. 13). Guidance on evaluating PRA technical adequacy is provided in Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 14). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decision Making: Technical Specifications," (Ref. 15), which includes CT changes as part of risk-informed decision making.

Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

3.1 Editorial Changes

The licensee proposed editorial changes which removes remnants of one-time only changes from some TSs which include:

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

These statements are obsolete and no longer meaningful, there is no technical justification to retain them in the CR-3 ITS, the removal of these footnotes have no impact on safety, and therefore, are acceptable.

3.2 Deterministic Evaluation of Extension of Allowed Outage Time to 7 Days

The Babcock and Wilcox Owners Group (B&WOG) conducted a study on the TS requirements of the LPI and RBS systems, and submitted a request for extending the AOT for these two systems (BAW-2295, Revision 1, October 1997) to 7 days. The B&WOG submittal is referred in this report as the “B&W submittal” or “the submittal”.

The deterministic evaluation consisted of a review of the plant systems and safety functions that are impacted by the entry into each AOT. The licensee quantitatively and qualitatively assessed the affected DHR, LPI and RBS safety functions. The licensee determined that there are no SSCs that will change status due to the changes. No new accidents or transients will be introduced by the proposed changes. 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 Final Safety Analysis Report (FSAR). Protective measures will be taken to ensure that unanticipated compromises to the 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 7 days for one inoperable train does not impact any assumptions or inputs in the FSAR.

In summary, FPC has requested to extend the AOT to 7 days for one train of RBS inoperable, one train of DC System inoperable, and one train of Decay Heat Seawater System inoperable. Add a new condition for one LPI subsystem inoperable with an AOT of 7 days, and add a new condition for one RBS train inoperable coincident with one containment cooling train inoperable with an AOT of 72 hours. The review by the B&WOG included both deterministic and probabilistic assessments. The changes made by FPC are consistent with this review. Since there are no changes in the inputs or assumptions in the FSAR, and the protections to maintain redundancy and diversity will remain intact, the NRC staff agrees with the assessments.

3.3 Deterministic Evaluation of Elimination of Second Completion Times Limiting Time

3.3.1 Background

Standard Technical Specifications

The use of the TS on second CT was based on an NRC staff concern that a plant could continue to operate indefinitely with an LCO governing safety significant systems never being met by alternately meeting the requirements of different conditions in the same specification. Some specifications allow entry into a condition and before the CT expires a different condition in the same specification is entered. The problem occurs when, previous to the expiration of the CT for the second condition, the first condition is entered for a second time, this process could allow an LCO to never be met. Multiple condition entry is permissible, but the repetitive entry into the same condition so that the LCO is not met for an extended period of time is unacceptable since TS conditions represent a temporary relaxation of the single failure criteria afforded by operable redundant safety systems. In order to overcome this issue, second CTs were used.

During development of STSs in 1991, the NRC staff could not identify any regulatory requirement or program which would prevent this misuse of the TSs described above. However, that is no longer the case. There are now two programs, the Maintenance Rule and the Reactor Oversight Process (ROP), which provide a strong disincentive to continue operation with concurrent multiple inoperabilities of the type the second CTs were designed to prevent. These regulatory processes discussed below, provide an equivalent level of plant safety without unnecessarily complicating some specifications by addition of a second CT for the LCO.

Maintenance Rule (10 CFR 50.65)

Issuance of the Maintenance Rule, in the early 1990's marked the advent of a regulation with significant implications for the evolution of TSs. Prior to 10 CFR 50.65, TSs were the primary rules governing operations, including what equipment must normally be in service, how long equipment can be out-of-service, compensatory actions, and surveillance testing to demonstrate equipment readiness. The goal of TSs is to provide adequate assurance of the availability and reliability of equipment needed to prevent, and if necessary mitigate, accidents and transients. The Maintenance Rule shares this same goal but operates at a more fundamental level with a dynamic and more comprehensive process. Thus, where the second CTs intent to prevent a repetitive entry into the same condition which is unacceptable may not have been an ideal process, 10 CFR 50.65 can also serve the purpose.

The Maintenance Rule assesses and manages inoperable equipment; however, the rule also considers all inoperable risk-significant equipment, not just the one or two systems governed by the same LCO. Under the TSs, the CT for one system within an LCO is not affected by inoperable equipment in another LCO. Therefore, the second CTs influenced the CT for one system based on the maintenance condition of another system, only if the two systems were required by the same LCO.

Under 10 CFR 50.65, 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," (Ref.16). RG 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Ref. 17). 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. Plant maintenance rule programs assure safe plant operation by managing plant configuration, thus augmenting the deterministic CTs in the TSs more successfully than implementing a second CT.

Also, the NRC resident inspectors monitor the licensee's corrective action process and could take action if the licensee's 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 identify poor maintenance practices that would result from multiple entries into the ACTIONS of the TSs which would contribute to unacceptable unavailability of these SSCs.

The Reactor Oversight Process

Satisfactory licensee performance in the cornerstone of mitigating systems provides reasonable assurance in monitoring the inappropriate use of condition CTs. The objective of this cornerstone is to monitor the availability, reliability, and capability of systems that mitigate the effects of initiating events to prevent core damage. Licensees reduce the likelihood of reactor accidents by maintaining the availability and reliability of mitigating systems. Mitigating systems include those systems associated with safety injection, decay heat removal, and their support systems, such as emergency alternating current (AC) power systems (which encompasses the AC Sources and Distribution System LCOs), and the Auxiliary Feedwater System. Inputs to the mitigating systems cornerstone include both inspection procedures and performance indicators to ensure that all safety objectives are being met.

Regulatory Information Summary 2001-11, "Voluntary Submission of Performance Indicator Data," endorses Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," which describes the tracking and reporting of performance indicators to support the NRC's ROP. Extended unavailability of these systems due to multiple entries into the required actions would affect the NRC's evaluation of the licensee's performance under the ROP.

NRC inspection findings for each plant are documented in inspection reports in accordance with Inspection Manual Chapter (IMC) 0612 and summarized in Plant Issues Matrices. Inspection findings are evaluated using the significance determination process in accordance with IMC 0609 to evaluate the safety significance of the findings.

Standard Technical Specifications Section 1.0, Use and Application

In addition to these programs, a paragraph was added to Section 1.3, "Completion Times," of CR-3's ITSs, stating that 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 should consider plant risk and shall limit the maximum contiguous time of failing to meet the LCO. This TS application helps provide an additional confidence level of plant safety without unnecessarily complicating some specifications by addition of a second CT for the LCO.

By letter dated February 25, 2008, the licensee made the following regulatory commitment:

> "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 for 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. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended. The administrative controls will ensure that Completion times is NOT extended beyond the additive Completion Times of the two Required Actions for restoration of OPERABLITY unless a risk evaluation is performed. If unit operation within an LCO will exceed the maximum

> Completion Time, then either the shutdown Condition within the LCO should be entered OR a risk evaluation shall be performed and the risk impact managed under CP-253, 'Power Operation Risk Assessment and Management.'"

3.3.2 Deterministic Evaluation

The NRC staff reviewed the proposed changes against the corresponding changes made to the STSs by TSTF-439, Revision 2, which the NRC staff has found to satisfy applicable regulatory requirements, as described above in Section 2.1. The staff determined that the proposed changes are consistent with NRC-approved TSTF-439, Revision 2. The following is the evaluation of those changes, and a conclusion of acceptability.

Proposed Changes

- The removal of the logical connector and the second CTs from the following:

    1. Condition A and B of Example 1.3-3, "Completion Times,"
    2. Condition A and C of ITS 3.6.6, "Reactor Building Spray and Containment Cooling Systems,"
    3. Condition A and B of ITS 3.7.5, "Emergency Feedwater (EFW) System,"
    4. Condition A and B of ITS 3.8.1, "AC Sources – Operating," and
    5. Condition A, B, and C of ITS 3.8.9, "Distribution Systems – Operating."

- The removal of text from ITS Section 1.3 discussing second CTs.

- 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 CTs are not inappropriately extended.

- The revision of the Bases of the affected section to reflect the changes mentioned above.

Technical Specification Example 1.3-3

Technical Specification Example 1.3-3 is revised to eliminate the second CTs for Required Actions A.1 and B.1 and to replace the discussion regarding second CTs with the following:

> "It is 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. 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."

The revised discussion addresses the concern the NRC staff had in 1991. The second CTs are being deleted because an equally effective solution to resolve the concern the NRC staff had in

1991 is now available with the Maintenance Rule and the ROP. Both the Maintenance Rule and the ROP address the issue in terms of comprehensively identifying specific equipment unavailability problems, and addressing these problems through the Corrective Actions Program and through compliance with TS CT limits.

Reactor Building Spray and Containment Cooling System

Technical Specification 3.6.6, “Reactor Building Spray and Containment Cooling Systems,” has a 72 hour CT for one RBS train inoperable (Condition A) and a 7-day CT for one required containment cooling train inoperable (Condition C). Condition A and C have a second CT of 10 days from the discovery of failure to meet the LCO. Restoring either one of the two inoperable conditions, i.e., either the inoperable RBS train for Condition A or the inoperable containment cooling train for Condition C, would result in exiting that condition. The second CT is limiting if multiple entries into and out of these conditions results in an indefinite period of time without meeting to the LCO. However, such frequent, repeated failures of the RBS and containment cooling systems would be readily identified by two independent programs, the Maintenance Rule and ROP, representing a strong disincentive to such operations.

Additionally, a licensing basis commitment regarding CR-3 contained in the licensee’s site procedure for the LCO tracking program requires that administrative controls ensure that CTs for those conditions are not inappropriately extended. As stated in Section 4.0 of this document, “The administrative controls will ensure that CT is NOT extended beyond the additive CTs of the two Required Actions for restoration of OPERABILITY unless a risk evaluation is performed.”

Emergency Feedwater System

Technical Specification 3.7.5, “Emergency Feedwater (EFW) System,” has a 7-day CT for one inoperable steam supply to a turbine driven EFW pump (Condition A) and a 72 hour CT for one EFW train inoperable for reasons other than Condition A (Condition B). Conditions A and B have a second CT of 10 days from discovery of failure to meet the LCO. Restoring either one of the two inoperable conditions, i.e. either the inoperable steam supply for Condition A or the inoperable EFW train for Condition B, would result in exiting that condition. The second CT is limiting if multiple entries into and out of these Conditions results in an indefinite period of time without meeting the LCO. However, such frequent, repeated failures of the EFW system would be readily identified by two independent programs, the Maintenance Rule and ROP, representing a strong disincentive to such operations.

Additionally, a licensing basis commitment regarding CR-3 contained in the licensee’s site procedure for the LCO tracking program requires that administrative controls ensure that CTs for those conditions are not inappropriately extended. As stated in Section 4.0 of this document, “The administrative controls will ensure that CT is NOT extended beyond the additive CTs of the two Required Actions for restoration of OPERABILITY unless a risk evaluation is performed.”

AC Sources - Operating

Technical Specification 3.8.1, “AC Sources - Operating,” has a 72 hour CT for one required offsite circuit inoperable (Condition A) and a 72 hour CT (or 14-day CT if alternate AC power is available) for one emergency diesel generator inoperable (Condition B). Both Condition A and Condition B have a second CT of “6 days from discovery of failure to meet the LCO” (or 17 days if from discovery of failure to meet LCO if alternate AC power is available). If Condition A or B is entered, and before that inoperable system is restored, the other Condition is entered, then Condition D applies, which is both Condition A and B inoperable, and plant operation is limited to 12 hours. Should either inoperable condition be restored, that condition and Condition D is exited. The second CT is limiting if repetitive entry into the previously restored conditions results in the LCO not being met for an extended period of time.

As stated above, the Maintenance Rule assesses and manages inoperable equipment, and the ROP monitors the availability of mitigating systems, including the emergency AC sources (emergency diesel generator unavailability). Such frequent, repeated failures of the AC sources would be reported to the NRC, representing a strong disincentive to such operations.

Additionally, a licensing basis commitment regarding CR-3 contained in the licensee's site procedure for the LCO tracking program requires that administrative controls ensure that CTs for those conditions are not inappropriately extended. As stated in Section 4.0 of this document, "The administrative controls will ensure that CT is NOT extended beyond the additive CTs of the two Required Actions for restoration of OPERABILITY unless a risk evaluation is performed."

Distribution Systems - Operating

Technical Specification 3.8.9, "Distribution Systems - Operating," has an 8-hour CT for one AC electrical power distribution subsystem inoperable (Condition A), and an 8-hour CT for one AC vital bus subsystem inoperable (Condition B), and a 2 hour CT for one direct current (DC) electrical power distribution subsystem inoperable (Condition C). Conditions A, B, and C have a second CT of 16 hours from discovery of failure to meet the LCO. The second CT limits plant operations from any potential AOT extensions if a condition in this LCO is entered, but before the CT for that condition is passed, a second different condition is entered; and again, before the CT for the second condition is passed, the first condition is entered again.

As previously mentioned, two supporting programs, The Maintenance Rule and the ROP provide the necessary assured safety that no LCOs will be abused.

Additionally, a licensing basis commitment regarding CR-3 contained in the licensee's site procedure for the LCO tracking program requires that administrative controls ensure that CTs for those conditions are not inappropriately extended. As stated in Section 4.0 of this document, "The administrative controls will ensure that CT is NOT extended beyond the additive CTs of the two Required Actions for restoration of OPERABILITY unless a risk evaluation is performed."

3.3.3 SUMMARY AND CONCLUSION

The NRC staff concludes that multiple continuous entries into conditions, without meeting the LCO, will be controlled by the licensee's configuration risk management programs, which were implemented to meet the requirements of the Maintenance Rule to assess and manage risk, and controlled by the Use and Application convention discussed in Section 1.3 of the TSs. The ROP, coupled with the Maintenance Rule, provide adequate assurance against inappropriate use of combinations of conditions that result in a single contiguous occurrence of failing to meet the LCO. Accordingly, consistent with TSTF-439, the NRC staff finds the proposed changes for CR-3 acceptable.

3.4 TECHNICAL EVALUATION - PROBABILISTIC RISK ASSESSMENT

3.4.1 Detailed Description of the Proposed Change in Accordance with TSTF-430

- TS 3.5.2, "ECCS – Operating." A new Condition A is added to address one LPI subsystem inoperable, with a 7-day CT. The existing Condition A is retained as Condition B with a 72-hour CT, and is modified with the addition of the words "for reasons other than Condition A".

- TS 3.6.6, "Reactor Building Spray and Containment Cooling Systems." Condition A is modified to change the CT from 72 hours to 7 days. A new Condition D is added to address the condition with one RBS and one containment cooling train inoperable, with a 72-hour CT. Existing Conditions D, E, and F are retained as Conditions E, F, and G, respectively.

- TS 3.7.8, "Decay Heat Closed Cycle Cooling Water (DC) System." Condition A is modified to change the CT from 72 hours to 7 days.

- TS 3.7.10, "Decay Heat Seawater System." Condition A is modified to change the CT from 72 hours to 7 days.

3.4.2 Review Methodology

In accordance with SRP Section 19.2 and Section 16.1, the staff reviewed the submittal using the three-tiered approach and the five key principles of risk-informed decision making presented in RG 1.174 and RG 1.177. The probabilistic risk evaluation of the licensee's proposed changes to the TSs using the three-tiered approach and addresses the Key Principle 5 of the five key principles outlined in RGs 1.174 and 1.177.

Key Principle 5: Performance Measurement Strategies - Implementation and Monitoring Program

RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms.

An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application.

3.4.3 Probabilistic Risk Assessment Technical Evaluation

The evaluation presented below addresses the staff's philosophy of risk-informed decision making, that when the proposed changes result in a change in CDF or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (Key Principle 4).

3.4.3.1 Tier 1: Probabilistic Risk Assessment Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk. The Tier 1 staff review involves two aspects: (1) evaluation of the validity of the CR-3 PRA models and their application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

Probabilistic Risk Assessment Quality

The objective of the PRA quality review is to determine whether the CR-3 PRA used in evaluating the proposed changes is of sufficient scope, level of detail, and technical adequacy for this application. The staff review evaluated the PRA quality information provided by the licensee in their submittal, including industry peer reviews results.

The CR-3 PRA model addresses internal events at power for both level 1 (core damage) and level two (containment performance and large early release). The PRA is an updated individual plant examination (IPE) model originally developed in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities", and associated supplements. Revisions to the model have been made to maintain consistency with plant design and operations. Administrative controls are in place for the updates to the PRA models, including written procedures and reviews. Computer software for processing probabilistic safety analysis (PSA) model inputs are verified and validated per administrative procedures.

The PRA has been peer reviewed using the nuclear industry peer certification review process in September 2001. The licensee identified that the peer reviewers were independent of the original developers of the model and were not company employees. The licensee identified significant changes made to the PRA model to address peer review issues, including:

- 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 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.
- Update of the level two analysis.

The licensee stated that all peer review items which affect the PRA model have been addressed and are reflected in the model used in their submittal.

The licensee identified that as of the date of their submittal, there were no outstanding plant changes which would require a change to the PRA model, and that there were no planned plant changes scheduled to be implemented prior the fall 2007 refueling outage which would require a change to the PRA model.

The licensee stated that the analysis results of the cutsets for CDF and LERF were reviewed in order to assure that the resulting accident sequences gave reasonable results for the various configurations examined for this study.

The licensee's risk analyses evaluated an assumed single entry into the extended CTs each year, and in response to a staff request for additional information (RAI) addressed non-concurrent entries into the extended CTs as well as increased unavailability of the affected components via additional risk analyses provided. The licensee analyses addressed the potential for increased probability of common cause failure (CCF) when the extended CT is entered due to corrective maintenance.

The licensee identified and justified truncation levels used to generate the cutsets for this analyses.

Based on review of the above information, the staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), and SRP Section 19.1, and that the quality of the CR-3 PRA is sufficient to support the risk evaluation for internal events provided by the licensee in support of the proposed license amendment.

Probabilistic Risk Assessment Results and Insights

The risk metrics for ΔCDF and ΔLERF for internal events were calculated by assuming an additional 96 hours of unavailability for each train of each affected system, and by assuming that the extended CT would be entered simultaneously for each of the four affected systems. Topical Report BAW-2295-A, which is the basis for TSTF-430 and this proposed change determined that a 13-hour increase in unavailability of each train per year was the appropriate change to evaluate. The licensee provided an updated analysis to address the 13-hour increase in unavailability and to address non-simultaneous entry into the extended CT for each of the four systems. Because these assumptions are consistent with the approved topical report BAW-2295-A and TSTF-430 bases, the staff used these risk results to assess the acceptability of the proposed changes to the CR-3 TSs.

The ICCDP and ICLERP for internal events were calculated by setting the train-specific maintenance event to 1.0 and resolving for CDF and LERF, then subtracting the baseline CDF and LERF and multiplying by the AOT duration. External events risk contribution is addressed separately.

The licensee's methodology is consistent with the guidance of RG 1.177, Section 2.3.4 and Section 2.4 and is, therefore, acceptable to the staff.

The results of the licensee analyses of internal events for increased unavailability of the four systems are shown in Table 1. The ΔCDF value is based on an increase of 13 hours in unavailability for each train of each of the four systems, and the ICCDP and ICLERP values are based on the 7-day CT duration. The ΔLERF value is taken from BAW-2295-A and was not updated based on the very low calculated value of ICLERP.

Table 1: Risk Results

Risk Metric	**Acceptance Guidance**	**CR-3 Result**
ΔCDF	< 1E-6 per year - very small RG 1.174	1.97E-7
ΔLERF	< 1E-7 per year - very small RG 1.174	9E-9
ICCDP	< 5E-7 - RG 1.177	6.0E-7
ICLERP	< 5E-8 - RG 1.177	4.03E-10

The risk impacts for the proposed changes to extend the CTs from 72 hours to 7 days for the LPI, RBS, DC and RW systems were found to be within the RG 1.174 acceptance guidelines for very small changes of less than 1E-6/year ΔCDF and 1E-7/year ΔLERF. The risk impact of each individual entry into the extended CT for 7 days was found to be within the RG 1.177 acceptance guideline of less than 5E-8 for ICLERP. The ICCDP of 6.0E-7 is only slightly above the guidance of RG 1.177 of 5.0E-7, and well below existing industry guidance for performing maintenance activities (NUMARC 93-01, Section 11, endorsed by the staff in RG 1.182). Further, based on the licensee's discussion of the anticipated application of the extended CT to simultaneously perform maintenance on multiple systems rather than sequential maintenance, the risk impacts for the proposed change are conservatively calculated.

Therefore, the staff finds that the licensee has satisfied the intent of RG 1.177 (Section 2.4), RG 1.174 (Sections 2.2.4 and 2.2.5), and SRP Section 19.2.

<u>External Events</u>

The licensee stated that the CR-3 PRA model is an internal events model which does not address internal fires, seismic events, or other external events. The risk impact of the proposed change from these events was, therefore, evaluated by the licensee, separately.

<u>Seismic, External Flooding, Transportation Events</u>

The licensee identified that seismic events, external flooding, and transportation events all have initiating event frequencies below 1.0E-6 per year, based on the CR-3 Individual Plant Examination for External Events (IPEEE). At this frequency, assuming a bounding conditional core damage probability of 1.0, over the 7-day extended CT the ICCDP impact would be:

$$(1.0\text{E-}6\ \text{yr}^{-1})\ \times\ (7\ \text{days})\ \times\ (\text{yr}/365\ \text{days})\ =\ 1.9\text{E-}8$$

This is more than an order of magnitude below the ICCDP due to internal events (6.0E-7), and is conservatively calculated by assuming 1) no mitigation credit, and 2) zero baseline risk for the incremental calculation. Therefore, the risk impact from these external events during the extended CT is not significant.

High Winds

The licensee identified three events related to high wind speeds from the CR-3 IPEEE. One event involves a tornado, which is assumed to directly result in core damage, with a frequency of 6.3E-8 per year. Since core damage is assumed to occur, the extended CT has no impact on risk from this event.

Two other events, tornadoes of a specified intensity (frequency 2.1E-5 per year per IPEEE Section 5.1.1) and wind damage to the EFW tank (frequency 6.5E-6 per year per IPEEE Section 5.1.2) were evaluated for impact from the extended CT. The ICCDPs for the most limiting train were found to be 6.3E-11 for the tornado event, and 1.6E-9 for the EFW tank damage event. These values are more than two orders of magnitude below the risk impact from internal events and are, therefore, not significant for this application.

Fire

The licensee performed a sensitivity study of fire risk during the extended CT period for CDF using the IPEEE fire analyses as a basis. Credit was assumed for suppression systems and fire wrap, and any equipment not protected by fire wrap was assumed to fail if suppression systems were failed. Fire scenarios for the cable spreading room and control room were excluded since both safety trains are simultaneously affected, and so the unavailability of one train due to an extended CT would not change the risk impact of these fires.

Three cases were evaluated: a baseline CDF with nominal equipment unavailabilities, and two cases with train A or train B equipment unavailable per the extended CT. An ICCDP was then calculated for the 7-day CT duration. The most limiting case involved train B equipment, with an ICCDP of 6.5E-7.

The licensee identified specific rooms which contribute to fire risk during the extended CT, and has committed to limit transient combustibles and provide periodic (identified as a nominal 1 hour interval) fire watches in these rooms during use of the extended CT (scheduled or emergent).

The licensee evaluated the cutsets for the fire scenarios and determined that spurious opening of a pressurizer power-operated relief valve (PORV) would be significant while the extended CT was in effect. Therefore, the licensee has committed to limit transient combustibles and provide periodic (identified as a nominal one hour interval) fire watches in rooms containing cables associated with the pressurizer PORV and PORV block valves during use of the extended CT.

The dominant contributors to LERF at CR-3 are bypass sequences such as ISLOCAs (which are assumed not mitigatable), or SGTRs, for which the LPI and RBS systems are not significant mitigating systems. Therefore, any increase in LERF due to fire during the extended CT will be very small.

Based on the conservatively bounding analysis results of external events discussed above, and the commitments to limit transient combustibles and establish periodic fire watches for important fire rooms, the staff finds that the licensee has satisfied the intent of RG 1.177 (Section 2.3.2 and Section 2.3.6), RG 1.174 (Section 2.2.3), and SRP Section 19.2.

Shutdown and Transition Risk

The licensee stated that the risk during shutdown is reduced by performing maintenance on the decay heat removal components while at power, since these components are the primary source of core cooling during shutdown conditions. This is conservatively not considered in calculating the risk impact of the proposed changes.

Uncertainty

The licensee's risk analyses are point estimates of the mean value. In response to an RAI, the licensee provided an estimate of the mean value accounting for parametric uncertainties in the data. The results demonstrated that the results are not particularly sensitive to the uncertainties in the underlying data, and the point estimate is, therefore, a reasonable approximation of the true mean values.

The licensee also provided an assessment of sources of uncertainty identified as important to this analysis. The following were quantitatively evaluated using sensitivity studies:

- Small LOCA Initiating Event Frequency – this is a dominant initiating event.
- Component Failure Rates – this addressed the specific components subject to the extended CT.
- Loss of Offsite Power Frequency – this is a generic industry issue impacted by grid aging and loading concerns.
- Assumption of Expected Unavailability – extending the CT may result in increased average annual unavailability of the affected systems.
- Assumption of Room Cooling Requirements – this plant-specific issue involves the assumed failure position of EFW valves when room cooling is lost affecting the valve control systems.

None of the sources of uncertainty evaluated were found to have a significant impact on the results of the analyses.

Common Cause Failure

The extended CT may be applied when the cause of entry into the CT is for emergent repairs or corrective maintenance. In such cases, there is increased potential that the opposite train may be unavailable due to a CCF. The licensee assessed the potential risk impact of this scenario by evaluating the CCF probability as conditional on the failure of the opposite train pump and reassessing the CDF and ICCDP. No credit was assumed for any equipment recovery.

The most limiting ICCDP is associated with corrective maintenance of raw water pump RWP-3B. The ICCDP is evaluated as 1.10E-6 over the 7-day extended CT. Unplanned maintenance unavailabilities are a small fraction of overall maintenance unavailability, and would not typically extend to the full 7-day limit. Further, the licensee identified that equipment failures or deficiencies are assessed to determine the potential for a CCF impact on other redundant components. Most potential CCF issues would likely be identified within 24 hours, and a different TS action requirement would then be applicable instead of the 7-day extended CT.

Therefore, the application of the extended CT for corrective maintenance does not result in an unacceptable risk increase.

3.4.3.2 Tier 2 - Avoidance of Risk-Significant Plant Configurations

The second tier requires a licensee to provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out-of-service in accordance with the proposed TS change.

The licensee identified compensatory measures applicable for maintenance activities planned which extend beyond the current 72-hour CT:

- Higher risk conditions will not be planned during an extended outage of the LPI, BS, DC or RW systems. Higher risk conditions are defined by the licensee as an "orange" color code involving an ICCDP of greater than 1E-5 over 7 days, or 1E-6 over 36 hours.
- No planned maintenance or discretionary equipment manipulations will be permitted on the Remote Shutdown Panel, Appendix R Cooler, and the opposite train of LPI, BS, DC, RW, EFW, Auxiliary Feedwater System, EFW Initiation and Control System, HPI, and their power supplies (AC and DC).
- Extended preventive maintenance outages will not be initiated if adverse weather conditions, as designated by emergency preparedness procedures, are anticipated.

Based on the above, the staff finds the licensee's Tier 2 evaluation of potential risk significant configurations and the proposed tier 2 restrictions support the implementation of changes to TS, and is acceptable to the staff.

3.4.3.3 Tier 3 - Risk-Informed Configuration Risk Management

The third tier requires a licensee to develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.

The licensee identified a commitment to implement a plant procedure (CP-253, "Power Operation Risk Assessment and Management"), which applies a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities, including emergent work or unplanned degradation of equipment. CR-3 will avoid entering an extended CT for LPI, BS, DC or RW systems, which would be "higher risk (Orange Color Code)", as discussed under tier 2 for planned activities.

Based on the licensee's conformance to the requirements of the guidelines of RG 1.177, the staff finds the licensee's Tier 3 program supports the proposed changes to TS, and is acceptable.

3.4.4 Comparison With Regulatory Guidance

The proposed changes to extend the CTs associated with LPI, BS, DC, and RW systems meets the acceptance guidance of RG 1.174, and the guidance outlined in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," and Section 16.1, "Risk-Informed Decision Making: Technical

Specifications," of the NRC's SRP, NUREG-0800. Further, the proposed changes are consistent with the guidance of RG 1.177.

3.4.5 Staff Findings and Conditions

The risk impacts for ΔCDF, ΔLERF, and ICLERP, as estimated by the licensee are within the acceptance guidelines for RGs 1.174 and 1.177 for the proposed changes to TS. The risk impact for ICCDP is reasonably consistent with the acceptance guidelines for RG 1.177. The licensee's Tier 2 analysis and commitments provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out-of-service in accordance with the proposed TS change. The licensee's Tier 3 CRMP was found to be consistent with the RG 1.177 CRMP guidelines.

3.4.6 Probabilistic Risk Evaluation Conclusion

The Tier 1 risk impacts for the proposed changes to TS to extend the CT for LPI, RBS, DC and RW systems is within the RG 1.174 and RG 1.177 acceptance guidelines for ΔCDF, ΔLERF, and ICLERP, and reasonably consistent with the RG 1.177 acceptance guidelines for ICCDP. The Tier 2 analysis provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out-of-service in accordance with the proposed TS change. The licensee's Tier 3 CRMP is consistent with the RG 1.177 CRMP guidelines. The proposed change to the TS satisfy the fourth key principle of risk-informed decision making identified in RG 1.174 and RG 1.177 and are therefore the NRC staff finds it acceptable for implementing risk-informed decision making.

4.0 REGULATORY COMMITMENT

In Reference 5, The NRC staff 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, RBS, DHCCC, or RW Systems planned to extend beyond the current 72 hours allowed in the TS, CR-3 has committed to 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 not plan any maintenance that results in "Higher Risk" (Orange Color Code) during an extended outage (greater than 72 hours) of the LPI, RBS, DHCCC, 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).
- CR-3 will not initiate an extended preventative maintenance outage (greater that 72 hours) on the LPI, RBS, DHCCC, or RW System if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.
- When extended maintenance (greater than 72 hours) is scheduled on a train of the LPI or RBS 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.

- When extended maintenance (greater than 72 hours) is scheduled on a train of the DHCCC or RW System, CR-3 will limit transient combustibles in the seawater room and establish a periodic fire watch in the seawater room.

The licensee made a regulatory commitment to implement these compensatory measures and these will be put in-place prior to implementing this revision. These proposed changes do not impact any assumptions and inputs in the safety analyses. The increased AOT will allow longer corrective maintenance to be completed at power, without requiring a plant shutdown. This proposal will reduce shutdowns due to a limiting condition for operation requirement.

The licensee in its letter dated April 13, 2007, and as supplemented by letters dated September 4 and September 13, 2007, and February 25, 2008 has committed to the following with regards to its proposed changes to the CR-3 FOL and TSs:

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-DC System.	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW-DC System
The Remote Shutdown Panel, the Appendix R Cooler and the opposite train of LPI, BS, DC, RW-DC, EFW, Auxiliary Feedwater System, Emergency Feedwater Initiation and Control System, HPI, and their power supplies (AC and DC) 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-DC System
CR-3 will not initiate an extended preventive maintenance outage (greater than 72 hours) on the LPI, BS, DC or RW-DC 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-DC System

Commitment	Due Date
When extended maintenance (greater than 72 hours) is performed (scheduled or emergent) on a train of the LPI or BS System, CR-3 will limit transient combustibles in, and establish a periodic fire watch in the decay heat pump vault of the opposite train, and the following rooms: • Non-safety 4160 V and 480 V Switchgear Rooms • Opposite train ES 4160 V and ES 480 V Switchgear Rooms • Opposite train battery room • Opposite train charger room • Opposite train Inverters room • Remote Shutdown Panel Room • Relay/CRD Room and Adjoining Corridor • 'B' EFIC Room • Cable Spreading Room	During extended (greater than 72 hours) outage on the LPI or BS System
When extended maintenance (greater than 72 hours) is performed (scheduled or emergent) on a train of the DC or RW-DC System, CR-3 will limit transient combustibles in, and establish a periodic fire watch in the seawater room, and in the following rooms: • Non-safety 4160 V and 480 V Switchgear Rooms • Opposite train ES 4160 V and ES 480 V Switchgear Rooms • Opposite train battery room • Opposite train charger room • Opposite train Inverters room • Remote Shutdown Panel Room • Relay/CRD Room and Adjoining Corridor • 'B' EFIC Room • Cable Spreading Room	During extended (greater than 72 hours) outage on the DC or RW-DC System
When extended maintenance (greater than 72 hours) is performed (scheduled or emergent) on a train of the LPI, BS, DC or RW-DC System, CR-3 will limit transient combustibles and establish a periodic fire watch in the fire zones containing routed cables associated with the pressurizer PORV and PORV Block Valves. These rooms include: • PORV/PORV Block Valve power supply breaker areas • Cable Spreading Room • Relay/CRD Room and Adjoining Corridor • Intermediate Building 119' elevation • Auxiliary Building 119' elevation • 'B' ES 4160 V Switchgear Room • Remote Shutdown Room • 'A'/'B' Battery Room	During extended (greater than 72 hours) outage on the DC LPI, BS, DC or RW-DC System

Commitment	Due Date
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 for 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. These administrative controls shall ensure that the CTs for those Conditions are not inappropriately extended. The administrative controls will ensure that CT is NOT extended beyond the additive CTs of the two Required Actions for restoration of OPERABILITY unless a risk evaluation is performed. If unit operation within an LCO will exceed the maximum CT, then either the shutdown Condition within the LCO should be entered OR a risk evaluation shall be performed and the risk impact managed under CP-253, "Power Operation Risk Assessment and Management."	This will be implemented in conjunction with the license amendment.

5.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

6.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 52167). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from D. E. Young to U. S. NRC, “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,” April 13, 2007 (Agencywide Documents Access Management System (ADAMS) Accession Number ML071060357).

2. Letter from J. A. Franke to U. S. NRC, “Crystal River Unit 3 - License Amendment Request #295, Revision 0: Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Response to Request For Additional Information (TAC No. MD5241),” September 4, 2007 (ADAMS Accession Number ML072490420).

3. Letter from D. E. Young to U. S. NRC, “Crystal River Unit 3 - License Amendment Request #295, Revision 0: Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Response to Request For Additional Information (TAC No. MD5241),” September 13, 2007 (ADAMS Accession Number ML072600153).

4. Letter from D. E. Young to U. S. NRC, “Crystal River Unit 3 - License Amendment Request #295, Revision 0: Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Response to Request For Additional Information (TAC No. MD5241),” February 25, 2008 (ADAMS Accession Number ML080640630).

5. TSTF-430, Rev. 2, “AOT Extension to 7 Days for LPI and Containment Spray (BAW-2995-A, Rev. 1),” (ADAMS Accession Number ML040050062).

6. TSTF-439, Rev. 2, “Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO,” (ADAMS Accession Number ML051860296).

7. NUREG-1430, Revision 3, “Standard Technical Specifications Babcock and Wilcox Plants,” June 2004.

9. NRC Letter from T.H. Boyce to TSTF, “Status of TSTF-439, ‘Eliminate Second Completion Times Limiting Time From Discovery of Failure To Meet an LCO’” (ADAMS Accession Number ML060120272), January 11, 2006.

10. BAW-2295-A, “Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems,” Revision 1, September 1999.

11. USNRC, Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1,” November 2002.

12. USNRC, Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,” August 1998.

13. NUREG-0800, Standard Review Plan 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007.

14. NUREG-0800, Standard Review Plan 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, June 2007.

15. USNRC, NUREG-0800, Standard Review Plan 16.1, "Risk-Informed Decision Making: Technical Specifications," Revision 1, March 2007.

16. USNRC, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000.

17. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", Revision 3, July 2000.

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