UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 13, 2015

Site Vice President
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: CHANGES TO TECHNICAL SPECIFICATION 3.1.3.4 REGARDING CONTROL ELEMENT ASSEMBLY DROP TIMES (CAC NO. MF6459)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 246 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (WF3). This amendment consists of changes to the Technical Specifications (TSs) and Final Safety Analysis Report (FSAR) in response to your application dated July 2, 2015, as supplemented by letters dated August 14, September 23, and October 8, 2015.

The amendment revises the limiting condition for operation (LCO) of TS 3.1.3.4 to increase the maximum individual control element assembly (CEA) drop time from the fully withdrawn position to 90 percent inserted from less than or equal to 3.2 seconds to less than or equal to 3.5 seconds and increases the maximum arithmetic average of all CEA drop times from less than or equal to 3.0 seconds to less than or equal to 3.2 seconds. Entergy Operations, Inc. (the licensee) also proposed to update the FSAR to account for the increase in the CEA drop times in TS LCO 3.1.3.4.

The NRC staff finds that the reanalysis supporting the proposed TS change uses the NRC-approved methodology and that the results of the reanalysis of the FSAR, Chapter 15, events, meet the applicable requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Therefore, the NRC staff concludes that the proposed TS LCO 3.1.3.4 and changes to the FSAR are acceptable.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a non-proprietary version of the SE that is provided in Enclosure 2.

NOTICE: Enclosure 3 to this letter contains Proprietary Information. Upon separation from Enclosure 3, this letter is DECONTROLLED.

A copy of our related SE is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

April L. Pulvirenti

April L. Pulvirenti, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 246 to NPF-38
2. Safety Evaluation (non-proprietary)
3. Safety Evaluation (proprietary)

cc w/encls 1 and 2: Distribution via Listserv

# ENCLOSURE 1

**AMENDMENT NO. 246**

**ENTERGY OPERATIONS, INC.**

**WATERFORD STEAM ELECTRIC STATION, UNIT 3**

**DOCKET NO. 50-382**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 246
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (EOI), dated July 2, 2015, as supplemented by letters dated August 14, September 23, and October 8, 2015, 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 Technical Specifications as indicated in the attachment to this license amendment, and the Facility Operating License No. NPF-38 is hereby amended, as addressed below:

Paragraph 2.C.2 of Facility Operating License No. NPF-38 is hereby amended to read as follows:

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 246, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Paragraph 2.C.20 of Facility Operating License No. NPF-38 is hereby amended to read as follows:

20. Control Element Assembly Drop Time Curve Validation (Amendment 246)

Prior to Cycle 21 Mode 2 operation, the licensee shall verify the control element assembly drop time test data demonstrates faster control element assembly drop times than the drop time curve provided in Table 15.0-5 of the Final Safety Analysis Report, as amended.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-38 and
Technical Specifications

Date of Issuance: November 13, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 246

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

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

Facility Operating License

| REMOVE | INSERT |
|---|---|
| -4- | -4- |
| -9- | -9- |
| -10- | -10- |

Technical Specifications

| REMOVE | INSERT |
|---|---|
| 3/4 1-23 | 3/4 1-23 |

or indirectly any control over (i) the facility, (ii) power or energy produced by the facility, or (iii) the licensees of the facility. Further, any rights acquired under this authorization may be exercised only in compliance with and subject to the requirements and restrictions of this operating license, the Atomic Energy Act of 1954, as amended, and the NRC's regulations. For purposes of this condition, the limitations of 10 CFR 50.81, as now in effect and as they may be subsequently amended, are fully applicable to the equity investors and any successors in interest to the equity investors, as long as the license for the facility remains in effect.

(b) Entergy Louisiana, LLC (or its designee) to notify the NRC in writing prior to any change in (i) the terms or conditions of any lease agreements executed as part of the above authorized financial transactions, (ii) any facility operating agreement involving a licensee that is in effect now or will be in effect in the future, or (iii) the existing property insurance coverages for the facility, that would materially alter the representations and conditions, set forth in the staff's Safety Evaluation enclosed to the NRC letter dated September 18, 1989. In addition, Entergy Louisiana, LLC or its designee is required to notify the NRC of any action by equity investors or successors in interest to Entergy Louisiana, LLC that may have an effect on the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

EOI is authorized to operate the facility at reactor core power levels not in excess of 3716 megawatts thermal (100% power) in accordance with the conditions specified herein.

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 246, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(a) The first performance of SR 6.5.17, in accordance with Specification 6.5.17.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from April 17, 2004, the date of the most recent successful tracer gas test, as stated in the October 8, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 6.5.17.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from April 17, 2004, the date of the most recent successful tracer gas test, as stated in the October 8, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 6.5.17 .d, shall be within 18 months, plus the 138 days allowed by SR 4.0.2, as measured from August 13, 2008, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

20. <u>Control Element Assembly Drop Time Curve Validation (Amendment 246)</u>

Prior to Cycle 21 Mode 2 operation, the licensee shall verify the control element assembly drop time test data demonstrates faster control element assembly drop times than the drop time curve provided in Table 15.0-5 of the Final Safety Analysis Report, as amended.

D. The facility requires an exemption from certain requirements of Appendices E and J to 10 CFR Part 50. These exemptions are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 10 (Section 6.1.2) and Supplement No. 8 (Section 6.2.6), respectively. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. These exemptions are, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. EOI 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 plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Physical Security, Safeguards Contingency and Training & Qualification Plan," and was submitted on October 4, 2004.

EOI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The EOI CSP was approved by License Amendment No. 234 and supplemented by a change approved by Amendment Nos. 239 and 241.

F. Except as otherwise provided in the Technical Specifications or the Environmental Protection Plan, EOI shall report any violations of the requirements contained in Section 2.C of this license in the following manner. Initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73(b), (c) and (e).

G. Entergy Louisiana, LLC shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as the date of issuance and shall expire at midnight on December 18, 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

original signed by H.R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. (DELETED)
2. Attachment 2
3. Appendix A (Technical Specifications) (NUREG-1117)
4. Appendix B (Environmental Protection Plan)
5. Appendix C (Antitrust Conditions)

Date of Issuance: March 16, 1985

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The arithmetic average of the CEA drop times of all CEAs from a fully withdrawn position, shall be less than or equal to 3.2 seconds; and the individual CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.5 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with:

a. Tavg greater than or equal to 520°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With either the average CEA drop time or any individual CEA drop time determined to exceed the above limits, restore the CEA drop time to within the above limits before proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At each refueling outage.

# ENCLOSURE 2

**SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION**

**RELATED TO AMENDMENT NO. 246 TO**

**FACILITY OPERATING LICENSE NO. NPF-38**

**ENTERGY OPERATIONS, INC.**

**WATERFORD STEAM ELECTRIC STATION, UNIT 3**

**DOCKET NO. 50-382**

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

**Redacted information is identified by blank space enclosed within double brackets.**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 246 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

## 1.0 INTRODUCTION

By application dated July 2, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15197A106), as supplemented by letters dated August 14, September 23, and October 8, 2015 (ADAMS Accession Nos. ML15226A346, ML15268A013, and ML15281A223, respectively), Entergy Operations, Inc. (Entergy or the licensee), requested changes to the Technical Specifications (TSs) for Waterford Steam Electric Station, Unit 3 (WF3). Portions of the letters dated July 2, and September 23, 2015, contain proprietary information and, accordingly, those portions have been withheld from public disclosure.

The supplements dated September 23, and October 8, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 8, 2015 (80 FR 53892).

The proposed changes would revise the limiting condition for operation (LCO) of TS 3.1.3.4 to increase the maximum individual control element assembly (CEA) drop time from the fully withdrawn position to 90 percent inserted from less than or equal to 3.2 seconds to less than or equal to 3.5 seconds and increase the maximum arithmetic average of all CEA drop times from less than or equal to 3.0 seconds to less than or equal to 3.2 seconds. The licensee also proposed to update the Final Safety Analysis Report (FSAR) to account for the CEA drop times increase in TS 3.1.3.4. The proposed changes were requested to address the concerns regarding the CEA drop time surveillance testing for Cycle 20, when WF3 experienced a challenge in meeting the requirements of TS 3.1.3.4 governing CEA drop time testing.

## 2.0 REGULATORY EVALUATION

The NRC's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TSs include items in five categories. These categories include (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls. The regulations in 10 CFR 50.36(d)(2)(ii) sets forth four criteria to be used in determining whether an LCO is required to be included in TSs. The following two criteria are applicable to the CEA drop times:

> Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
>
> Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

General Design Criterion (GDC) 10 in Appendix A to 10 CFR Part 50, "Reactor design," requires that "the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits [SAFDLs] are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

GDC 15, "Reactor coolant system design," requires that "the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

## 3.0 TECHNICAL EVALUATION

### 3.1 Background

#### 3.1.1 Historical Drop Time Tests

The LCO of TS 3.1.3.4 for WF3 specifies the maximum permitted CEA drop times. Surveillance Requirement 4.1.3.4 requires that the CEA drop time be demonstrated through measurement prior to reactor criticality. Measuring drop time is required for all CEAs following each removal and reinstallation of the reactor vessel head for specifically affected individual CEAs following any maintenance or modification to the CEA drive system, which could affect the drop time of those specific CEAs, and at each refueling outage. Should the average CEA drop time or an individual CEA drop time exceed the LCO limits of TS 3.1.3.4, the plant is not allowed to proceed to power operation (Mode 1) or startup (Mode 2).

For the SR 4.1.3.4 test prior to Cycle 19 startup, the CEA drop times noticeably increased and approached the TS 3.1.3.4 LCO limits. For Cycle 20, the CEA drop times increased even further above Cycle 19 and exceeded the LCO requirements for CEA drop times. The licensee repeated the test and the CEA drop times passed the TS limit. For Cycle 15, Cycle 16, and Cycle 17, the data showed that the measured drop times were lower, approximately the same amongst those three cycles, and within the LCO limits. No data was presented for Cycle 18.

The licensee provided information regarding the measurement methods for Cycle 20-1 and Cycle 20-2 SR 4.1.3.4 testing and stated that no changes were made between the performance of the Cycle 20-1 and Cycle 20-2 CEA drop time tests. The CEA drop time test software records all the CEA positions every 50 milliseconds. The CEA drop time test software will always round time up to the next 50 millisecond interval, ensuring that the recorded rod drop times will be conservative, resulting in the CEA drop time test software having a CEA drop time uncertainty of 100 milliseconds (0.1 seconds). The Cycle 20-1 initial test was performed with the average time of 3.024 seconds. The Cycle 20-2 second test was performed with the average time of 2.967 seconds. The difference between the Cycle 20-1 and Cycle 20-2 test results was within the CEA drop time test software uncertainty band.

3.1.2 Apparent and Potential Causes for the Increased CEA Drop Times

The licensee indicated that the apparent cause of the increase in the CEA drop times is the combined effects of major plant modifications, including increased pressure drop through the fuel assemblies as a result of the transition to next generation fuel (NGF) assemblies; increased core flow as a result of increased cross-sectional area of the replacement steam generator (RSG) U-tubes; and lower pressure drop in the RSGs that results in increased flow through the CEA guide tubes. The major plant modifications made during the plant cycles are provided below:

- Appendix K Power Uprate was implemented in Cycle 12.
- Extended Power Uprate was implemented in Cycle 14.
- Alternative Source Term was implemented in Cycle 14.
- A partial core of NGF was implemented in Cycle 16 and a full core of NGF was implemented in Cycle 17.
- CEA Replacement was made between Cycle 17 and Cycle 18.
- Steam Generator Replacement was made between Cycle 18 and Cycle 19.
- Reactor Head Replacement was made between Cycle 18 and Cycle 19.
- Control Element Drive Mechanism Replacement was made between Cycle 18 and Cycle 19.

The licensee also identified the following two potential causes: 1) a change in control element drive mechanisms (CEDM) voltage decay time between the original and replacement CEDMs causing a delay in the start of the CEA drop; and 2) the change in as-built manufacturing tolerances in the replacement reactor pressure vessel head (RPVH) resulting in increased friction between the CEA extension shaft and the RPVH and, thus, slower CEA motion. The licensee indicated that both the apparent and potential causes are due to one-time plant modifications and, therefore, the CEA drop times are not expected to further degrade over time.

The NRC staff reviewed the apparent and potential causes listed by the licensee as one-time changes that could reasonably be expected to result in slower CEA drop times, but makes no finding on the root cause(s) of the slower CEA drop times. The requirements in Criteria 2 and 3 of 10 CFR 50.36(d)(2)(ii) result in the inclusion of CEA drop times in the TS, but no drop time numerical values are specified in any NRC regulation. GDCs 10 and 15 only require that the CEA drop times provide appropriate margin to assure that the SAFDLs and reactor coolant system (RCS) design conditions are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Given that there are no regulations that provide specific numerical CEA drop times, the CEA drop times that the licensee proposes to include in the TS will be considered acceptable if they provide appropriate margin to assure that the SAFDLs and RCS design conditions are not exceeded and are accounted for in the FSAR, Chapter 15, and TSs/SRs.

### 3.2 Proposed TS 3.1.3.4

Since failure to pass the CEA drop time test required by TS LCO 3.1.3.4 would preclude entering either Mode 1 or Mode 2 operations, the licensee proposed a modification to TS LCO 3.1.3.4 to increase the required limits for the CEA drop times to maintain plant operations for Cycle 21 and beyond. Specifically, the proposed changes to TS LCO 3.1.3.4 are: (1) the maximum allowable individual CEA drop time from the fully withdrawn position to 90 percent inserted is increased from less than or equal to 3.2 seconds to less than or equal to 3.5 seconds; and (2) the maximum allowable arithmetic average of all CEA drop times is increased from less than or equal to 3.0 seconds to less than or equal to 3.2 seconds for plant conditions with average reactor coolant system temperature greater than or equal to 520 degrees Fahrenheit (F) and all reactor coolant pumps (RCPs) operating. In support of the proposed TS changes, the licensee's application provided the technical analyses, including reanalysis and evaluation of the events discussed in Chapter 15, "Accident Analyses," of the WF3 FSAR.

The NRC staff reviewed the proposed TS change and associated technical analyses provided by the licensee against the applicable regulations and guidance documents. Specifically, the NRC staff reviewed the submittal to assure that the licensee's analysis considered all applicable events discussed in Chapter 15 of the WF3 FSAR; used the adequate analytical methods, plant conditions, and assumptions in meeting applicable acceptance criteria; and that the proposed TS values for the CEA drop times adequately reflect the results of the reanalysis.

### 3.3 Events Re-analyzed with the Proposed CEA Drop Times

The CEA insertion (or rod drop) during a reactor trip adds negative reactivity into the core and to shut down the unit during transients or accidents, some of which are discussed in Chapter 15 of the WF3 FSAR. The licensee assessed all of the FSAR, Chapter 15, transient and accident analyses to determine the impact of the proposed CEA drop times by classifying the FSAR, Chapter 15, events into the following four categories:

1. Events for which a reactor trip does not occur or was not credited;
2. Events for which consequences are not sensitive to the CEA drop time change;
3. Events are bounded by other Chapter 15 events; and
4. Events for which consequences are affected by the CEA drop time change.

In its application dated July 2, 2015, the licensee discussed the bases for the classification of each event. For the first three categories, no reanalysis was required. The licensee identified 14 events to be placed in Category 4, including:

1. Increased Main Steam Flow
2. Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
3. Steam System Piping Failures: Post-Trip Return-To-Power
4. Steam System Piping Failures: Pre-Trip Power Excursion Analysis
5. Loss of Condenser Vacuum
6. Loss of Normal Feedwater Flow
7. Feedwater System Pipe Breaks
8. Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System
9. Total Loss of Forced Reactor Coolant Flow
10. Single RCP Shaft Seizure/Sheared Shaft
11. Uncontrolled CEA Withdrawal from a Subcritical Condition
12. Uncontrolled CEA Withdrawal at Low Power
13. CEA Ejection
14. Asymmetric Steam Generator Transient

The licensee performed a reanalysis of the Category 4 events affected by the revised CEA drop times. The NRC staff reviewed the summary of event categories and associated bases for classification of each event. Based on its review, the NRC staff finds that all of the applicable FSAR Chapter 15 events were considered and correctly classified into the four categories by the licensee.

3.3.1 Methods and Plant Initial Conditions

3.3.1.1 Analysis Methodology

Since the NRC approval of the WF3 FSAR, Chapter 15, analysis for the extended power uprate by License Amendment No. 199, dated April 15, 2005 (ADAMS Accession No. ML051030082), and alternative source term implementation by License Amendment No. 198, dated March 29, 2005 (ADAMS Accession No. ML050890248), the licensee implemented the following two major plant changes: the use of the NGF and the installation of the RSG. These changes were addressed by various license amendment requests (LARs) listed in Table 4.0-2 of the

July 2, 2015, submittal and by the 10 CFR 50.59, "Changes, tests, and experiments," process. The NRC-approved LARs relevant to the use of the NGF listed in Table 4.0-2 are:

- License Amendment No. 200 (ADAMS Accession No. ML051290368) that included ZIRLO™ in TS 5.3.1 as an acceptable fuel rod cladding and allowed the use of Westinghouse physical code package (PHOENIX/ANC, PARADON);
- License Amendment No. 210 (ADAMS Accession No. ML061930421) that allowed the use of a Westinghouse methodology in support of the use of Zirconium diboride burnable absorber coating on fuel pellets;
- License Amendment No. 214 (ADAMS Accession No. ML080880014) that allowed the use of the emergency core cooling system performance analysis that supports the implementation of 16x16 NGF;
- License Amendment No. 215 (ADAMS Accession No. ML080380005) that allowed optimized ZIRLO™ as an acceptable fuel cladding material; and
- License Amendment No. 224 (ADAMS Accession No. ML092880237) that allowed a new departure from nucleate boiling ratio (DNBR) safety limit for 16x16 NGF.

License Amendment No. 225 (ADAMS Accession No. ML093480507) is the NRC-approved LAR for the use of RSG in regards to the transient and accident analysis. License Amendment No. 225 allowed the relocation of the SG level-high trip requirement from the TS to the Technical Requirements Manual.

The Westinghouse reload analysis methodology was applied to the analysis supporting the use of NGF and RSG. The licensee stated that this methodology was consistent with the methodology described in the FSAR that was approved for use by the NRC. The current analysis (i.e., analysis of record (AOR)) was incorporated into the FSAR, Chapter 15, and used this same methodology. The reanalysis with the proposed CEA drop times started with the AOR. Based on the fact that the reanalysis used the methodology previously approved by the NRC, the NRC staff concludes that the methods used in the reanalysis for the transients and accidents are acceptable.

3.3.1.2 Input Parameters

In comparison with the AOR, all reanalysis cases used the new CEA drop time curve. In addition, several other input parameters were modified to partially offset the loss of DNBR margin due to the revised CEA drop time curve. The additional modified input parameters included the axial shape index (ASI), fuel temperature coefficient, and initial thermal margin.

3.3.1.2.1 Average CEA Drop Time Methodology and Associated Drop Times

The revised CEA drop time curve used in the reanalysis is shown in Table 4.0-1 of the July 2, 2015, submittal. It assumes a 0.6 second delay between trip breaker opening and when the magnetic flux of the CEA holding coils decay enough to allow the CEAs to fall, and another 2.6 seconds (versus 2.4 seconds used in the AOR) for actual CEA insertion from beginning of movement to 90 percent insertion. This timing corresponds to the proposed TS LCO 3.1.3.4 value of less than or equal to 3.2 seconds (versus less than or equal to 3.0 seconds in the current TS) between when the trip breakers open until the CEAs are 90 percent inserted. The licensee stated that the revised rod drop position versus time was based on plant data and a conservative analysis approach. The plant data revealed that the time for the rods to reach 90 percent inserted reveals that for Cycles 5 through 17, the average drop time was 2,800 milliseconds, while the average drop time for Cycles 19 and 20 was 2,950 milliseconds. To bound the change in average drop times, the rod drop time used in the revised analysis was increased by 200 milliseconds, or 0.2 seconds, for all rod positions from 10 percent to 90 percent inserted. To maintain continuity in the revised drop time curve, the 5 percent insertion position time was increased by 0.15 seconds. The shape of the revised drop time curve is the same as the one utilized in the current FSAR, Chapter 15, Non-LOCA (loss-of-coolant accident) safety analysis, but shifted to the right (slightly slower) to account for the impact of major plant modifications, as discussed in Section 2.1 of this safety evaluation. Therefore, the shape of the rod drop time curve remains constant in the revised analyses supporting the TS LCO 3.1.3.4 change request. The NRC staff reviewed the revised drop time curve used in the reanalysis and finds that it results in slower addition of negative reactivity due to CEA insertion. The revised rod drop time curve will be verified as part of the Cycle 21 rod drop testing, as required by the license condition discussed in Section 3.3 of this safety evaluation. Based on the above information, the NRC staff concludes that the revised CEA drop time curve is acceptable for use in the reanalysis.

The licensee's reanalysis of the 14 applicable events assumed all holding coils released their CEAs at the same time and all CEAs fall at the same rate. The reactivity change due to CEA insertion is characterized by the average CEA drop time. The use of the average CEA drop time methodology was previously approved by the NRC in License Amendment No. 58 (ADAMS Accession No. ML021760257) with two conditions:

(1) Any fuel management change that significantly affects the core wide axial or radial power profiles, such as axial blankets or ultra-low leakage fuel management, may necessitate reverification of the average CEA drop time analysis.

(2) Changes that would significantly affect the CEA drop time distribution, such as changes to the CEDM circuits, large increases in the core flow pressure drop, changes in the total drop weight of the CEAs or changes in the location of the CEAs, may also require reverification of the average CEA drop time concept.

Condition (1) addresses the possibility of CEAs in peripheral assemblies inserting faster than the average insertion speed, but not contributing as much to the overall reactivity inserted, due to the very low power peripheral assemblies in certain fuel management plans. In addressing the compliance with Condition (1), the licensee stated that WF3 has 87 full length CEAs, and the CEAs at peripheral locations with one assembly corner in the vicinity of the shroud often faces high-power fresh fuel. The current WF3 fuel management plan locates high-power fresh fuel with peripheral CEAs and would not significantly affect the core wide axial or radial power profiles. Based on the above information, the NRC staff finds that the WF3 fuel management plan adequately addresses Condition (1).

In addressing compliance with Condition (2) regarding the change of the CEA drop time distribution, the licensee provided the measured data for the individual CEA drop times for Cycle 15 through Cycle 20 in Figure 2 of the July 2, 2015, submittal. The measured data shows that the overall drop times for Cycle 19 and Cycle 20 were about 0.15 seconds greater than those measured for Cycles 15 through 17. Cycle 17 was the last Cycle prior to implementation of plant changes including CEA replacement, steam generator replacement, reactor head replacement and control element drive mechanism replacement. The distribution of the CEA drop times is consistent from Cycle 15 to Cycle 20 and remains somewhat random from cycle to cycle, confirming that the locations of the CEAs with increased drop times do not concentrate in certain regions of the core. The NRC staff finds that the random distribution of the CEA drop times indicates that the plant modifications did not affect the CEA drop time distribution and the reverification of the average CEA drop time concept by the licensee is not necessary; therefore, the licensee has satisfied Condition (2).

Based on the licensee's compliance with two conditions, the NRC staff concludes that the previously approved average CEA drop time methodology remains acceptable for use in the reanalysis to support the proposed changes to TS LCO 3.1.3.4.

3.3.1.2.2 Axial Shape Index

In the reanalysis of certain events, the ASI was reduced to partially offset the loss of DNBR margin due to the change to the CEA drop time curve. When required, the analysis margin for the axial power distribution was reduced from an ASI of +0.3 to +0.2. The licensee stated that +0.2 was used because specific analyses may use different ASI values, provided that they are conservative and provide bounded results. The NRC staff reviewed the ASI of +0.2 used in the reanalysis and finds that it bounds the ASI upper limit of +0.16, as specified in Section 3.2.7 of the "Core Operating Limits Report [COLR] – Cycle 20 Revision 1," dated July 29, 2014 (ADAMS Accession No. ML14241A013), for WF3. Additionally, the ASI upper limit of +0.16 is for power levels greater than 50 percent and the applicable reanalysis is based on an initial power level of 100 percent. Based on the above information that an ASI of +0.2 bounds the ASI upper limit of +0.16 in the COLR, the NRC staff concludes that the use of an ASI value of +0.2 is acceptable.

An ASI of +0.2 was used in the event reanalysis discussed in Sections 3.3.2.1, 3.3.2.2, 3.3.2.9, 3.3.2.10 and 3.3.2.14 of this safety evaluation.

3.3.1.2.3 Fuel Temperature Coefficient

The least negative fuel temperature coefficient (FTC) was changed from -0.00113 $\Delta\rho/\sqrt{K}$ to -0.0013 $\Delta\rho/\sqrt{K}$. The licensee provided the bases for the least negative FTC and justified the adequacy of the use of the new value. The FTC ranges from the most negative (-0.0026 $\Delta\rho/\sqrt{K}$) to the least negative (-0.0013 $\Delta\rho/\sqrt{K}$) for the current WF3 operating fuel cycles and these bounding cycle independent values are confirmed every reload cycle. A conservative value of -0.00113 $\Delta\rho/\sqrt{K}$ was selected during the WF3 extended power update analyses with the intention of bounding a wider range of subsequent reload cycles. As part of the reanalysis to support the revised CEA drop time curve, this extra analysis conservatism was removed and a value of -0.0013 $\Delta\rho/\sqrt{K}$ was used. The NRC staff reviewed the FTC change and finds that an FTC of -0.0013 $\Delta\rho/\sqrt{K}$ is the least negative value for a range applicable to the WF3 operating fuel cycles. Because the FTC is the least negative value, which will be confirmed by the licensee every fuel cycle as required by the existing methodology and analysis in the WF3 licensing basis, the NRC staff concludes its use in the reanalysis is acceptable.

The FTC value of 0.0013 $\Delta\rho/\sqrt{K}$ was used in the event reanalysis discussed in Sections 3.3.2.1, 3.3.2.2, 3.3.2.9, and 3.3.2.10 of this safety evaluation.

3.3.1.2.4 Initial Thermal Margin

For the reanalysis that used additional initial thermal margin, the WF3 Core Operating Limit Supervisor System (COLSS) already contains additional initial thermal margin, but it was not utilized in the AOR. For the asymmetric steam generator transient, the AOR initial thermal margin, defined as the Required Over-Power Margin (ROPM), is 119.9 percent and the revised CEA drop time reanalysis ROPM is 120.23 percent. The ROPM contained in the COLSS is 123 percent. The NRC staff finds that the reanalysis used additional initial thermal margin, but is within the initial thermal margin contained within the COLSS. Based on this information, the ROPM used in the reanalysis is less than the ROPM contained in the COLSS, therefore, the NRC staff concludes that the revised ROPM values are acceptable.

The initial thermal margin was changed in the event reanalysis discussed in Sections 3.3.2.2 and 3.3.2.14 of this safety evaluation.

3.3.2 Results of the Event Reanalysis

The NRC staff has reviewed the results of the reanalysis for the 14 affected events listed in Section 3.3 of this safety evaluation and each event is evaluated in subsections 3.3.2.1 through 3.3.2.14 below.

3.3.2.1 Increased Main Steam Flow

The WF3 FSAR, Section 15.1.2.3, "Increased Main Steam Flow with a Concurrent Loss of Offsite Power," describes the increased main steam flow with a concurrent loss of offsite power event. The revised evaluation started with the AOR and incorporated the following changes, as described in more detail in Section 3.3.1.2 of this safety evaluation: 1) revised CEA drop time, 2) ASI limits, and 3) the least negative FTC. The changes to the ASI and FTC were made to partially offset the loss of DNBR margin, provided the change to the CEA drop time curve.

The WF3 FSAR, Section 15.1.2.3.3.1, "Mathematical Model," states that two different analysis models, a typical case and a worst departure from nucleate boiling (DNB) performance case, were used for this event. The licensee stated that only the worst DNB performance case was reanalyzed. The NRC staff finds this acceptable because it is more conservative in terms of the fuel failure rate versus the typical case.

The NRC staff reviewed the results from the increased main steam flow with a concurrent loss of offsite power reanalysis. The results from the reanalysis show that the minimum DNBR was reduced slightly from 1.071 to 1.051. The calculated fuel failure rate increased from less than 4.0 percent to less than 5.0 percent and is below the acceptance criterion of less than 8 percent used as the fuel failure rate limit in the WF3 FSAR, Section 15.1.2.3. The increased main steam flow with a concurrent loss of offsite power event is not limiting with respect to peak linear heat rate. The increase in peak secondary pressure is based on the loss of condenser vacuum (LOCV) results that shows an increase in peak secondary pressure of less than 1 pound per square inch (psi) and remains well below the acceptance criteron of 110 percent of the secondary side design pressure. The LOCV results were used for the worst overpressurization case because the limiting events with respect to peak primary and secondary pressures are the LOCV and feedwater line break accidents. These two events are limiting because they have the closest approach to the TS 2.1.2, "Reactor Coolant System Pressure," safety limit of 2,750 pounds per square inch absolute (psia) and secondary pressure limit of 1,210 psia (110 percent of the secondary side design pressure).

The increased CEA drop time impacts the power reduction and slightly increases the amount of energy deposited into the primary coolant system post-trip. The reanalysis used the LOCV event because it produced the largest post-trip primary and secondary pressure spike due to the loss of secondary heat removal capability. Since the LOCV event produces a more severe pressure transient, the slight energy deposition increase would be expected to have the most significant impact on this event. Thus, applying the LOCV event results to the increased main steam flow event ensured that the most conservative case was used in the reanalysis.

The licensee used the loss of normal feedwater flow (LONF) event (i.e., a loss of feedwater due to pump or valve failures, not a feedwater line break) to evaluate the transient characteristics (e.g., energy deposition and associated steam releases) of the increased main steam flow with a concurrent loss of offsite power event. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases beyond full CEA rod insertion, the impact of the revised CEA drop time becomes zero and the radiological releases due to steam release and break flow would remain unchanged. This event assumes an 8 percent failed fuel limit and the results demonstrate that the fuel failure limits remain unchanged, indicating that the radiological source terms remain the same. Thus, there is no change to the radiological results for this event. Additionally, using the results of the LONF analysis is adequate for all of the events discussed in the submittal because the deposited energy and associated steam releases may be different between events, but the energy and steam changes for the 0.2 second CEA drop time increase are small. This NRC staff position is applicable to Sections 3.3.2.1 through 3.3.2.14 of this safety evaluation with respect to the evaluation of the radiological releases.

Based on its review, the NRC staff finds that the reanalysis for the increased main steam flow event, with a concurrent loss of offsite power, used the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.2 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

The WF3 FSAR, Section 15.1.2.4, "Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component, "describes the inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component event. The revised analysis started with the AOR and incorporated the changed input parameters described in Section 3.3.1.2 of this safety evaluation.

The NRC staff reviewed the results from the inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component reanalysis. The analysis results show that the minimum DNBR increased slightly from 1.247 to 1.277, which continues to meet the acceptance criterion of greater than or equal to 1.24 provided in TS 2.1.1.1, "DNBR" and in the WF3 FSAR, Sections 4.3A.4.1, "DNBR Analysis," and 4.3A.4.2, "Effects of Fuel Rod Bowing on DNB Margin." The increase in peak primary pressure was based on the LOCV results that showed an increase in peak primary pressure of less than 1 psi. The peak primary pressure of 2,584 psia remains below the TS 2.1.2 safety limit of less than 2,750 psia, thereby meeting the requirements in GDC 15. The increase in peak secondary pressure was based on the LOCV results that show an increase in peak secondary pressure of less than 1 psi. The peak secondary pressure of 1,118 psia remains below the acceptance criterion of less than 1,210 psia (110 percent of the secondary side design pressure). The licensee used the LOCV event because the limiting events, with respect to peak primary and secondary pressures, are the LOCV and feedwater line break accidents. LOCV and feedwater line break accidents are the limiting events because they have the closest approach to the TS 2.1.2 safety limit of 2,750 psia and secondary pressure limit of 1,210 psia.

The increased CEA drop time impacts the power reduction and slightly increases the amount of energy deposited into the primary coolant system post-trip. The reanalysis used the LOCV event because it produced the largest post-trip primary and secondary pressure spike due to the loss of secondary heat removal capability. Since the LOCV event produces a more severe pressure transient, the slight energy deposition increase would be expected to have the most significant impact on this event. Thus, applying the LOCV event results to the inadvertent opening of a steam generator atmospheric dump valve event ensured that the most conservative case was used in the reanalysis.

The LONF event was used to evaluate the transient characteristics (e.g., energy deposition and associated steam releases) that would be applicable to the inadvertent opening of a steam generator atmospheric dump valve analysis. The analysis shows that the differences in primary and secondary system energy after a reactor trip is insignificant. As the time increases beyond a full CEA rod insertion, the impact of the revised CEA drop time becomes zero and the radiological releases due to steam release and break flow remain unchanged. Thus, the radiological consequences are not impacted by the change in the CEA drop time.

Based on its review, the NRC staff finds that the reanalysis for the event, an inadvertent opening of a steam generator atmospheric dump valve event with a concurrent single failure of an active component, uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.3 Steam System Piping Failures Post-Trip Return-To-Power

The post-trip return to power (R-t-P) conditions occur when the negative reactivity from a reactor trip CEA insertion and the emergency core cooling system boron addition is not sufficient to offset the core reactivity increase due to the moderator temperature coefficient (MTC) reactivity feedback from cooldown. The WF3 FSAR, Section 15.1.3.1, "Steam System Piping Failures Post-Trip Return-to Power," discusses the analysis of four main steam line break (MSLB) events with the maximum potential for a post-trip R-t-P. The events are:

- A guillotine break MSLB at hot-full power (HFP) with loss of offsite power (LOOP).
- A guillotine break MSLB at HFP with offsite power available.
- A guillotine break MSLB at hot-zero power (HZP) with LOOP.
- A guillotine break MSLB at HZP with offsite power available.

In addition, the above combinations were analyzed for both inside containment (IC) and outside containment (OC) break locations.

The licensee indicated that the consequences of the post-trip R-t-P MSLB events were dominated by (1) the total CEA insertion reactivity added, (2) the rate of RCS cooldown, and (3) the RCS moderator and fuel reactivity feedback. The NRC staff finds that the licensee's discussion for the MSLB reactivity response is consistent with the FSAR sequence of the post-trip R-t-P MSLB events. As discussed in the WF3 FSAR, Section 15.1.3.1, during an MSLB event, the increase in steam flow from a pipe break in the main steam system initiates an increase in energy removal by the affected steam generator from the RCS. The excess energy removal resulted in a reduction of the reactor coolant temperature and pressure. With presence of a negative MTC, the cooldown resulted in an increase in core reactivity.

The NRC staff reviewed the information in the WF3 FSAR, Section 15.1.3.1, for the four MSLB events analyzed with the maximum potential for an R-t-P. The steam generator blowdown rate is dependent on the steam generator pressure; the licensee discussed the impact of the revised CEA drop time on the steam generator pressure. The initial steam generator pressures are based upon the beginning operating conditions. The post steam line break steam generator pressures fall together until a main steam isolation signal actuates and the main steam isolation valves (MSIVs) close. Once the MSIVs close, the affected steam generator continues to blowdown, whereas the unaffected steam generator pressure recovers or maintains. The affected steam generator blowdown duration is dominated by the steam generator water inventory, initial temperatures, and break size. The increased CEA drop time impacts the power reduction post-trip and the amount of energy deposited into the primary coolant system. The slightly longer drop time results in a slight increase in the amount of energy added to the RCS; however, it would have a minimal impact on the steam generator pressure and no change to the event characteristics or consequences would occur with respect to the R-t-P.

The revised CEA drop time would not produce any significant effects on the transient characteristics; therefore, the total reactivity added to the core during MSLB events would not change. Based on this information, the reactivity addition during an MSLB remains unchanged with the new CEA drop time, and the NRC staff finds that the AOR regarding maximum post-trip fission power, DNBR, and linear heat rate remains valid and meets the applicable acceptance criteria. Therefore, the NRC staff concludes that the AOR is acceptable for supporting the proposed TS change.

3.3.2.4 Steam System Piping Failures: Pre-Trip Power Excursion Analysis

The WF3 FSAR, Section 15.1.3.3, "Steam System Piping Failures: Pre-Trip Power Excursion Analysis," discusses the analysis for two MSLB events that were analyzed to maximize the potential for a pre-trip power excursion. The events are:

- An IC guillotine break MSLB at HFP with LOOP.
- An OC guillotine break MSLB at HFP with LOOP.

The WF3 FSAR, Section 15.1.3.3.3.3, "Results," discusses the results of the analysis for both events and indicates that the minimum DNBRs were predicted to occur at the time between the initiation of the CEA drop and completion of 90 percent insertion. The analysis assumed that a loss of offsite power triggered a coastdown of all of the RCPs, resulting in reduced RCS flow rates through the reactor core. When the CEA drop time increased from the current value to the proposed value, the power reduction would be delayed due to the slower CEA insertion time. For the core conditions with a delayed power level reduction and the same RCP coastdown flow rate as the AOR, the minimum calculated DNBRs would decrease.

The licensee reanalyzed both of the pre-trip MSLB cases based on the same initial plant conditions and assumptions as the AOR, but with the only change being the revised CEA drop time. The results of the reanalysis show that although the DNBR margin was reduced by approximately 33 percent, the minimum DNBR remained above the DNBR safety limit of 1.24. Also, since the peak core power occurred before the initiation of the CEA drop, the revised CEA drop time had no effect on the maximum peak linear heat rate (PLHR), assuring that the acceptable maximum PLHR in the AOR remained valid. The minimum DNBR value and the maximum PLHR demonstrate that the SAFDLs are met and no fuel failures occur.

The WF3 FSAR, Section 15.1.3.3.5.1, "Design Basis – Method of Analysis," describes the OC pre-trip MSLB radiological consequences. Section 15.1.3.3.5.1 states that the radiological consequences were bounded by the results calculated for a feedwater line break (FLB) due to both the modeling of releases for the FLB event and that no fuel failures (0 percent) were predicted for these two events. Section 15.1.3.3.5.1.1 of the WF3 FSAR discusses the IC pre-trip MSLB radiological consequences that were based on an assumption of 10 percent failed fuel. The reanalysis for the pre-trip MSLBs showed that no fuel failure occurred during the OC and IC pre-trip MSLB, assuring that the radiological source terms remained unchanged.

Radiological doses were calculated over a 2-hour and 8-hour period, and/or until shutdown cooling conditions were reached. The LONF event was reanalyzed to identify the effect of the CEA drop time changes on the transient characteristics over a long term period with respect to energy deposition and associated steam releases for the pre-trip MSLB event. In the September 23, 2015, supplement, the licensee provided additional justification for the use of the LONF results to support radiological releases. The LONF analysis shows that the differences in primary and secondary system energy after a reactor trip are insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become negligible, and the radiological releases due to steam release would remain the same.

The NRC staff reviewed the results of the LONF analysis applied to the two MSLB event reanalysis. For the LONF event, the licensee provided the results of the analysis in Figure 15-1 in its supplement dated September 23, 2015, showing the energy added to the RCS from time of trip (43.7 seconds) to 50 seconds for both the AOR case and the case with the revised CEA drop time, and the results in Figure 15-2 showing the energy deposition from the time of trip to 1,800 seconds for both cases. The deposited energy between the reactor trip time and 1,800 seconds is 161,485.7 megawatt thermal (MWt)-sec for the AOR and 162,228.1 MWt-sec for the revised CEA drop time analysis. This equates to an energy change of 0.46 percent, which would have a negligible change on the radiological consequences. Since the radiological source terms and steam releases used to determine the dose releases were not affected by the CEA drop time change, the NRC staff finds that the radiological releases in the AOR for the pre-trip MSLB event remains valid.

Based on its review, the NRC staff finds that the reanalysis for the pre-trip MSLB event uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.5 Loss of Condenser Vacuum

The WF3 FSAR, Section 15.2.1.3, "Loss of Condenser Vacuum," describes the LOCV event. The reanalysis modifies the AOR only with the revised CEA drop times and the results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only a minimal increase in peak primary pressure. The NRC staff reviewed the results from the LOCV reanalysis. The peak primary pressure increased from 2,711 to 2,712 psia. The peak secondary side pressure remained the same at 1,180 psia. The minimum DNBR and maximum peak linear heat rate remain within the acceptance criteria for the AOR, as discussed in the WF3 FSAR, Section 15.2.1.3. The licensee also analyzed a LOCV event with a failure of the pressurizer level control system. The results showed that sufficient time (minimum of 15 minutes after the reactor trip, as identified in FSAR Section 15.2.1.3) exists for operator action to mitigate the LOCV event prior to water covering the primary safety valve inlet nozzles.

Based on its review, the NRC staff finds that the reanalysis for the LOCV uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.6 Loss of Normal Feedwater Flow

The WF3 FSAR, Section 15.2.2.5, "Loss of Normal Feedwater Flow," describes the LONF event. The reanalysis modifies the AOR with only the revised CEA drop time. The NRC staff reviewed the results from the LONF event that demonstrate that the peak primary and secondary pressures remain below the acceptance criteria. The peak primary pressure increased from 2,248 psia to 2,268 psia, which is below the acceptance criterion of less than 2,750 psia, thereby meeting the requirements of GDC 15. The peak secondary pressure increased from 1,071 psia to 1,072 psia, which is below the acceptance criterion of less than 1,210 psia. The minimum DNBR and maximum peak LHR remain within the acceptance criteria.

Based on its review, the NRC staff finds that the reanalysis for the LONF event uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meets the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.7 Feedwater System Pipe Breaks

The WF3 FSAR, Section 15.2.3.1, "Feedwater System Pipe Breaks," describes the large and small feedwater system pipe break (FSPB) events. The reanalysis modifies the AOR with only the revised CEA drop time for both large and small FSPB cases. The NRC staff reviewed the reanalysis of the large and small FSPB events that demonstrated that the peak primary and secondary pressures remain below the acceptance criteria, with only a minimal increase in peak primary pressure, and that the minimum DNBR and maximum peak linear heat rate remain within the acceptance criteria. Additionally, the 575 gallons per minute of water that the emergency feedwater system provides to each steam generator remains sufficient to prevent dryout of the unaffected steam generator.

The licensee uses the LONF event to evaluate the transient characteristics with respect to energy deposition and associated steam releases that are applicable to the feedwater system pipe break analysis. Thus, the radiological consequences are not impacted by the change in CEA drop time, as discussed in Section 3.3.2.1 of this safety evaluation.

Based on its review, the NRC staff finds that the reanalysis for the FSPB events uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.8 Loss of Normal Feedwater Flow with a Single Failure

The WF3 FSAR, Section 15.2.3.2, "Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System," describes the LONF event with an active failure in the steam bypass system. The reanalysis modifies the AOR with only the revised CEA drop time. The NRC staff reviewed the results of the analysis that showed that the peak primary pressure increased from 2,171 psia to 2,191 psia, which remains below the acceptance criteria of less than 2,750 psia, thereby meeting the requirements of GDC 15. The peak secondary pressure increased from 1,053 psia to 1,054 psia, which remains below the acceptance criteria of less than 1,210 psia. Both the minimum DNBR and maximum peak linear heat rate remain within the acceptance criteria.

The licensee also reanalyzed a LONF event, plus a single failure of the pressurizer level control system, as described in the WF3 FSAR, Section 15.2.2.5.4.3. The reanalysis results show only minor differences in system parameters at 30 minutes (approximately 29 minutes after reactor trip), such that the transient differences are negligible.

Based on its review, the NRC staff finds that the reanalysis for the LONF event with an active failure in the steam bypass system uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.9 Total Loss of Forced Reactor Coolant Flow

The WF3 FSAR, Section 15.3.2.1, "Total Loss of Forced Reactor Coolant Flow," discusses the total loss of forced reactor coolant flow (LOF) reanalysis. The reanalysis results indicated that the peak primary pressure occurred within 5 seconds following the CEA insertion and the minimum DNBR occurred at the time between the initiation of the CEA drop and completion of 90 percent insertion. When the CEA drop time increased from the current value to the proposed value, the power reduction will be delayed due to the slower CEA insertion time and the energy generated would increase. For the core conditions with a delayed decrease in the power level and a reducing RCS flow rate, the minimum calculated DNBRs would decrease and the peak primary pressure would increase.

The NRC staff reviewed the LOF reanalysis. The licensee reanalyzed the LOF event using the same initial plant conditions and assumptions as the AOR, except for the following changes: (1) the CEA drop time that was revised to be consistent with the proposed TS value, (2) the ASI was reduced from +0.3 to +0.2 that was within the allowed upper limit of +0.16 required in the COLR, and (3) the least negative fuel temperature coefficient was revised to reflect the proposed fuel management plan. The results of the reanalysis showed that the minimum DNBR was reduced from 1.302 to 1.293, but remained above the DNBR safety limit of 1.24.

The licensee applied the LONF long-term results to the LOF event to determine the peak primary pressure. The licensee justified this approach by stating that the reanalysis of the LONF event showed that the proposed increase in the CEA drop time resulted in (1) a primary pressure increase of less than 20 psi and (2) a peak primary pressure less than the opening pressure of the pressurizer safety valves (PSVs). The licensee applied the LONF event to provide peak primary pressure results for those events that do not exceed the PSV setpoints. Since the AOR for the LOF event in WF3 FSAR, Table 15.3-1, showed that the peak primary pressure was less than the PSV opening setpoints, the licensee applied the LONF results to the LOF event to determine the peak primary pressure. The reanalysis with the increased CEA drop time showed a peak primary pressure increase of 20 psi (from 2,395 psia to 2,415 psia). The NRC staff finds that the calculated peak primary pressure of 2,415 psia is acceptable because the primary pressure increase of 20 psi credited in the peak pressure determination is small, the pressure margin of 335 psi (2,415 psia versus the pressure limit of 2,750 psia) is sufficient to compensate for uncertainties associated with the credited peak pressure increase, and it meets the GDC 15 requirements.

Based on its review, the NRC staff finds that the LOF reanalysis uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.10 Single Reactor Coolant Pump (RCP) Shaft Seizure/Sheared Shaft

WF3 FSAR, Section 15.3.3.1, "Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft," describes the single RCP shaft seizure/sheared shaft events. The AOR results indicated that at 1.50 seconds after the seizure/sheared shaft, the turbine/generator was tripped, resulting in (1) a loss of offsite power (2) a coast down of the remaining three RCPs, and (3) a further decrease in the reactor coolant flow. FSAR Table 15.3-3 shows that the peak primary pressure occurred within 5 seconds following the completion of the CEA insertion and the minimum DNBR occurred in the time between the initiation of the CEA drop and completion of 90 percent insertion. When the CEA drop time increased from the current value to the proposed value, the power reduction would be delayed due to the slower CEA insertion time and the energy generated in the core would increase. For the core conditions with a delayed power level reduction and the same RCP coastdown rates as the AOR, the calculated minimum DNBRs would decrease and the peak primary pressure would increase.

The licensee reanalyzed the RCP shaft seizure/sheared shaft events using the same initial plant conditions and assumptions as the AOR, except for the following changes: (1) the CEA drop time that was revised to be consistent with the proposed TS value, (2) the ASI was reduced from +0.3 to +0.2, which is within the allowed upper limit of 0.16 required in the COLR, and (3) the least negative fuel temperature coefficient is revised to reflect the proposed fuel management. The results of the reanalysis shows that the minimum DNBR was reduced from 1.1450 to 1.131. The calculated fuel failures increased from less than 8.0 percent to less than 10.5 percent, which were below the 15 percent assumed in the FSAR radiological analysis.

The LONF event was chosen by the licensee to evaluate the transient characteristics with respect to energy deposition and associated steam releases that would be applicable to the RCP shaft seizure/sheared shaft analysis. Thus, the radiological consequences were not impacted by the change in CEA drop time, as discussed in Section 3.3.2.1 of this safety evaluation.

The NRC staff reviewed the RCP shaft seizure/sheared shaft reanalysis results. The peak primary pressure increased by 20 psi to 2,442 psia and the peak secondary pressure increased by 1 psi to 1,118 psia, but was within the secondary side pressure limits of 110 percent of the design pressure. The increase in the peak primary pressure was based on the LONF long-term results and the increase in the peak secondary pressure was based on the LOCV results. In the supplement dated September 23, 2015, the licensee provided justification for the use of the results from the two different events, LONF and LOCV, to derive the peak primary and secondary pressure, respectively, for the RCP shaft seizure/sheared shaft events. For the increased CEA drop time, the peak primary pressure for the LONF event increased less than 20 psi and the peak primary pressure is less than the opening pressure of the PSVs. The licensee applied the LONF event to the shaft seizure/sheared shaft reanalysis to provide peak primary pressure results because it did not exceed the PSV setpoints. The peak primary pressure increased by 20 psi from 2,422 psia to 2,442 psia for the shaft seizure/sheared shaft event. The NRC staff determined that the calculated peak primary pressure of 2,442 psia is acceptable because the primary pressure increase of 20 psi credited in the peak pressure determination is small, the margin of 308 psi (2,442 psia versus the pressure limit of 2,750 psia) is sufficient to compensate for uncertainties associated with the credited peak pressure increase, and it meets the GDC 15 requirements.

As for the peak secondary pressure, the LOCV analysis increase of less than 1 psi was used. The NRC staff finds that the calculated peak secondary pressure of 1,118 psia is acceptable because the LOCV is the limiting peak secondary pressure event, and the calculated pressure is within the 1,210 psia acceptance criteria of 110 percent design pressure.

Based on its review, the NRC staff finds that the analysis of the RCP shaft seizure/sheared shaft event used the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.11 Uncontrolled CEA Withdrawal from a Subcritical Condition

The WF3 FSAR, Section 15.4.1.1, "Uncontrolled CEA Withdrawal from Subcritical Conditions," describes the uncontrolled CEA withdrawal from a subcritical condition event. The AOR results showed that at approximately 593 seconds into the transient, a reactor trip actuates on high logarithmic power and the CEAs started dropping into the core at 594 seconds. The WF3 FSAR, Table 15.4-1, shows that the minimum DNBRs were predicted to occur at the time between the initiation of the CEA drop and 90 percent insertion. When the CEA drop time increased from the current value to the proposed value, the power reduction would be delayed due to the slower CEA insertion time. For the core conditions with a delayed reduction in the power level, the minimum calculated DNBRs would decrease.

The NRC staff reviewed the reanalysis results. The licensee reanalyzed the uncontrolled CEA withdrawal from a subcritical condition event based on the same initial plant conditions and assumptions as the AOR, but with the proposed CEA drop time. The results of the reanalysis shows that (1) the minimum DNBR reduced from 4.31 to 3.66, which remained above the DNBR safety limit of 1.24 and (2) the maximum peak fuel centerline temperature of less than 3,500 degrees F remained within the acceptance criterion of 4,663 degrees F[1]. The results demonstrated that that SAFDL limits were met and no fuel failures occurred, thereby meeting the requirements of GDC 10.

The peak primary and secondary pressure increases of the LOCV event would bound the expected pressures for the uncontrolled CEA withdrawal from a subcritical condition event, thus, the primary and secondary pressures for the event remained within the pressure safety limits, thereby, meeting the requirements of GDC 15.

Based on its review, the NRC staff finds that the reanalysis uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.12 Uncontrolled CEA Withdrawal at Low Power

The WF3 FSAR, Section 15.4.1.2, "Uncontrolled CEA Withdrawal from Low Power Conditions," describes the uncontrolled CEA withdrawal at low power event. The result shows that at approximately 24.2 seconds into the transient, a reactor trip actuated on high variable

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[1] In the July 2, 2015, submittal, the licensee stated that the maximum peak fuel centerline temperature limit is 4,663 degrees F, but the FSAR Section 15.4.1.1.3 incorrectly states that it is 4,900 degrees F. The licensee stated that the correction will be made in the FSAR.

overpower, and the CEAs began dropping into the core at 25.3 seconds. The WF3 FSAR, Table 15.4-3, indicated that the minimum DNBRs were predicted to occur at the time between the initiation of the CEA drop and 90 percent insertion. When the CEA drop time increased from the current value to the proposed value, the power reduction would be delayed due to the slower CEA insertion. For the core conditions with a delayed reduction in the power level, the minimum calculated DNBRs would decrease.

The NRC staff reviewed the CEA withdrawal at low power reanalysis results. The licensee reanalyzed the CEA withdrawal at low power event based on the same initial plant conditions and assumptions as the AOR, but with the proposed slower CEA drop time. The results of the reanalysis shows that (1) the minimum DNBR reduced from 3.44 to 2.90, which remained above the DNBR safety limit of 1.24 and (2) the maximum peak fuel centerline temperature of less than 3,500 degrees F remained within the acceptance criterion of 4,663 degrees F. The results demonstrated that that SAFDL limits were met and no fuel failures occurred, thereby meeting the requirements of GDC 10.

The peak primary and secondary pressure increases of the LOCV event would bound the expected pressures for the CEA withdrawal from at low power event, thus, the primary and secondary pressures for the CEA withdrawal from at low power event remained within the pressure safety limits, thereby, meeting the requirements of GDC 15.

Based on its review, the NRC staff finds that the analysis of the uncontrolled CEA withdrawal at low power event uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis was acceptable for supporting the proposed TS change.

3.3.2.13 CEA Ejection

The WF3 FSAR, Section 15.4.3.2, "Control Element Assembly (CEA) Ejection," describes the analysis for the CEA ejection event. The licensee reanalyzed the CEA ejection event based on the same initial plant conditions and assumptions as the AOR, but with the proposed CEA drop time. The NRC staff reviewed the CEA ejection reanalysis results. The results of the reanalysis show that: 1) the peak primary pressure increases slightly and remains below the reactor coolant pressure boundary limits; 2) the total calculated fuel failures that experienced DNB clad damage increases from 11.6 percent to 12.1 percent of the total number of the fuel rods in the core, which remains below the value of 15 percent assumed in the FSAR radiological releases analysis; and 3) the maximum fuel rod radial average and maximum incipient centerline melting enthalpy remains within the acceptance criteria.

The LONF event was chosen by the licensee to evaluate the transient characteristics with respect to energy deposition and associated steam releases, which would be applicable to the CEA ejection analysis. Thus, the radiological consequences were not impacted by the change in CEA drop time, as discussed in Section 3.3.2.1 of this safety evaluation.

Based on its review, the NRC staff finds that the analysis of the CEA ejection event uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.14 Asymmetric Steam Generator Transient

The WF3 FSAR, Section 15.9.1.1, "Asymmetric Steam Generator Transient," describes asymmetric steam generator transients (ASGT). These transients result from the malfunction of one steam generator and are analyzed to determine the initial safety limit margins that must be maintained by the TSs LCOs such that, in conjunction with the Reactor Protection System Core Protection Calculator high differential cold leg temperature reactor trip, the DNBR and Fuel Centerline Melt (CTM) limits are not exceeded. The four cases examined in the FSAR include:

- Loss of Load to One Steam Generator (LL/1SG)
- Excess Load to One Steam Generator (EL/1SG)
- Loss of Feedwater to One Steam Generator (LF/1SG)
- Excess Feedwater to One Steam Generator (EF/1SG)

In the supplement dated September 23, 2015, the licensee stated that the limiting case for the ASGT analysis is the closure of the MSIV to steam generator #2 (LL/1SG) and only the limiting case was evaluated with the revised CEA drop time. The NRC staff finds that re-evaluating only the limiting case is acceptable because the results from this limiting case bound the remaining cases.

The revised evaluation started with the AOR and incorporated the following changes, which are described in more detail in Section 3.3.1.2 of this safety evaluation: (1) revised CEA drop time, (2) axial shape index limits, and (3) initial thermal margin. The changes to the ASI and initial thermal margin were made to partially offset the loss of DNBR margin provided the change to the CEA drop time curve.

The NRC staff reviewed the ASGT reanalysis results. The ASGT reanalysis demonstrates that the initial margins were adequate to ensure that the ASGT event does not violate the DNBR (greater than or equal to 1.24) and CTM (4,663 degrees F) fuel design limits, thereby meeting the requirements of GDC 10. The ASGT reanalysis demonstrated that the limits on maximum primary (2,750 psia) and secondary pressure (1,210 psia) were not exceeded, thereby meeting the requirements of GDC 15.

Based on its review, the NRC staff finds that the analysis of the ASGT events uses the approved methodology and proposed CEA drop time, and the results of the reanalysis meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the reanalysis is acceptable for supporting the proposed TS change.

3.3.2.15 Summary of Events for Which Consequences were Affected by the CEA Drop Time Change

The NRC staff reviewed the 14 events for which the consequences were affected by the CEA drop time change. The NRC staff finds that the licensee appropriately used the analysis methods of the AOR and that the results of the reanalysis with the inclusion of the 0.2 second drop time change were within the acceptance limits and met GDCs 10 and 15. The NRC staff concludes that the reanalysis of the 14 events are acceptable for supporting the proposed TS change.

3.3.3 Control Element Drive Mechanism Voltage Decay Time

As described in Section 2.1 of this safety evaluation, the licensee identified one potential cause for the increased CEA rod drop time to be a change in CEDMs voltage decay time between the original and replacement CEDMs. While considered less likely, this failure mode could not be ruled out and was investigated further by the licensee. The licensee originally reanalyzed the 14 events, as described in Section 3.3.2 of this safety evaluation, with a 0.6 second CEA holding coil decay time; however, this issue was not identified until after most of the reanalysis was completed.

To investigate the effects of a longer CEA holding coil decay time, the licensee provided an analysis of the limiting accident events using a 0.8 second CEA holding coil decay time. Three limiting events were selected for re-evaluation to bound the expected impact of the CEA drop time change due to the increased (i.e., 0.8 second) CEA holding coil decay time. By using the limiting events, the licensee stated that the other FSAR events can be evaluated and dispositioned under the 10 CFR 50.59 process with no adverse impact to the conclusions contained in its submittal dated July 2, 2015. The following limiting events were chosen with respect to peak primary and secondary pressure, minimum DNBR, and fuel failure results:

- LOCV
  - represents the limiting peak primary and secondary pressure events
- Sheared Shaft/Seized Rotor
  - produces the most rapid thermal degradation for this class of events
  - violates the DNBR SAFDL and results in fuel failure
- CEA Ejection
  - limiting event with respect to reactivity insertion
  - results in violating the DNBR SAFDL and results in fuel failure

The above three events were reanalyzed with the beginning of the CEA drop time curve shifted an additional 0.2 seconds (difference between 0.6 and 0.8 second decay time) from the originally proposed curve, but with the total drop time remaining the same (3.2 seconds). The NRC staff finds this 0.2 second shift acceptable because it assumes that the entire delay in the

CEA drop time is attributed to a delay in the start of CEA movement and expects that this will result in a more conservative analysis result.

The NRC staff's review of the results of the three reanalyses using the 0.8 second CEA holding coil delay time is provided below:

3.3.3.1 Loss of Condenser Vacuum

The licensee's analysis demonstrates the change in peak primary and secondary side pressure results for the LOCV event modeling a 0.8 second CEA holding coil delay. The LOCV reanalysis results indicate that the difference between this reanalysis (which examines the impact of the increased CEA holding coil decay), and the previous reanalysis presented in Section 3.3.2.5 of this safety evaluation, is negligible.

3.3.3.2 Sheared Shaft/Seized Rotor

The licensee's analysis demonstrates the change in fuel failure and minimum DNBR for the sheared shaft/seized rotor event modeling a 0.8 second CEA holding coil delay. The sheared shaft/seized rotor event is not limiting with respect to peak primary and secondary pressures. The results show that there are no changes in the minimum DNBR or calculated fuel failure between this reanalysis and the previous reanalysis presented in Section 3.3.2.10 of this safety evaluation.

3.3.3.3 CEA Ejection

The licensee's analysis demonstrates the change in fuel failure, fuel rod radial average enthalpy, and fuel rod centerline enthalpy results for the CEA ejection event modeling a 0.8 second CEA holding coil delay. The CEA ejection event is not limiting with respect to peak primary and secondary pressures. The calculated fuel failure increased from less than or equal to 12.1 percent to less than or equal to 12.4 percent, which remains within the acceptance criteria of less than or equal to 15 percent. The CEA ejection results indicate that the impact of the increased CEA holding coil decay on the CEA ejection analysis is minimal when compared to the previous reanalysis presented in Section 3.3.2.13 of this safety evaluation.

3.3.3.4 Control Element Drive Mechanism Voltage Decay Time Conclusion

The NRC staff reviewed the revised LOCV, Shared shaft/seized rotor, and CEA ejection analyses and finds that the reanalysis with 0.8 second voltage decay time demonstrate that the changes in (1) peak primary pressure, (2) peak secondary pressure, (3) minimum DNBR, (4) fuel rod radial average temperature, and (5) fuel rod centerline enthalpy results are negligible or minimal. All of the results remain within the acceptance criteria. Therefore, the NRC staff concludes that results of the reanalysis discussed in Section 3.3.2 of this safety evaluation remain valid even if the CEA holding coil decay time were to increase to 0.8 seconds.

3.3.4 Fuel Thermal Conductivity Degradation Analysis

The licensee describes the application of a penalty on allowable rod power to account for fuel thermal conductivity degradation (TCD). The exposure-dependent fuel temperature adjustment is listed in Table 1 of Attachment 9 to the July 2, 2015, submittal and its impact on allowable rod power is depicted in Figures 2 and 3 of the July 2, 2015, submittal.

The licensee stated that the NRC-approved FATES3B model was used to assess the fuel rods' performance over their irradiation history. However, FATES3B does not model the degradation in fuel thermal conductivity with increasing fuel rod exposure.

In 2011, Florida Power and Light (FPL) submitted an extended power uprate LAR for St. Lucie Plant, Unit 2 (ADAMS Accession No. ML110730116), which also relied upon FATES3B calculations. During the review, the NRC staff questioned the continued use of FATES3B and its lack of an exposure-dependent fuel thermal conductivity model. FPL stated that the FATES3B model contained other conservatisms that resulted in acceptable centerline temperature predictions and, consequently, conservative pellet average temperatures and stored energy estimates for the design and safety applications. To address the NRC staff's concerns, Westinghouse performed FATES3B benchmark calculations against the Halden Reactor Project (HRP) fuel temperature data. An exposure-dependent bias was revealed based upon this benchmark. As a result, FPL and Westinghouse proposed an [[ ]]. Using the NRC's fuel thermal-mechanical model FRAPCON-3.4 (ADAMS Accession No. ML11101A005), the NRC staff performed independent calculations that confirmed the adequacy of the FATES3B calculations and [[ ]] for the FPL extended power uprate LAR (ADAMS Accession No. ML12235A463).

According to the licensee, FATES3B calculations for WF3 will apply an [[ ]] equivalent to the TCD temperature adjustment applied in the FPL extended power uprate license amendment request. Due to differences in fuel rod design, the magnitude of the [[ ]] will change relative to the curve used by FPL. The licensee provided examples of the application of the [[ ]] to existing WF3 [[ ]].

Based upon the NRC staff's prior review of the FATES3B benchmark calculations against HRP high burnup fuel temperature data, the independent FRAPCON-3.4 confirmatory calculations, and considering that the Waterford TCD methodology follows FPL's methodology, the NRC staff concludes that the proposed [[ ]] is acceptable to account for TCD and is acceptable for supporting the proposed TS change.

3.4 License Condition

The analysis discussed in Sections 3.3.2 through 3.3.4 of this safety evaluation used the CEA drop time curves, as provided in the supplement dated August 14, 2015. These CEA drop time curves were derived from the CEA drop time curve used in the AOR with inclusion of the proposed 0.2 second increase. The licensee has not demonstrated that these CEA drop time curves are conservative with respect to the CEA drop time test data. To assure reactor operation remains within the analysis, as discussed in Sections 3.3.2 through 3.3.4 of this safety evaluation, the licensee should verify that the drop time data demonstrates faster CEA drop times than the provided drop time curve before power operations commence. Regulatory Commitment 2 intended to complete this validation 60 days after power operations began, as described in Section 4.0 of this safety evaluation. To prevent power operations in a condition outside of the analysis as discussed in Sections 3.3.2 through 3.3.4 of this safety evaluation, the licensee submitted the October 8, 2015, supplement that contained the following license condition to elevate modified Regulatory Commitment 2 to an obligation.

New License Condition 2.C.20 will state:

> Prior to Cycle 21 Mode 2 operation, the licensee shall verify the control element assembly drop time test data demonstrates faster control element assembly drop times than the drop time curve provided in Table 15.0-5 of the Final Safety Analysis Report, as amended.

Additionally, the drop time curve from the August 14, 2015, supplement was incorporated into Table 15.0-5 of the FSAR and the associated FSAR pages were submitted with the supplement dated October 8, 2015.

The proposed license condition states that the licensee will verify that the drop time test data for Cycle 21 of operation only. The licensee provided information that both the apparent and potential causes are due to one-time plant modifications and therefore the CEA insertion times are not expected to further degrade over time. TS LCO 3.1.3.4 will continue to assure that the CEA drop times meet the WF3 FSAR Chapter 15 analyses. The NRC staff finds that since the CEA insertion times are not expected to further degrade and that TS LCO 3.1.3.4 ensures CEA drop times meet the FSAR Chapter 15 analyses, a one-time review of the data after the Cycle 21 test is sufficient to ensure that the insertion curves are bounded by the insertion curves in Table 15.0-5 of the FSAR, as amended.

Based on the above, the NRC staff concludes that the proposed license condition will allow power operations only if the CEA drop time test data is bounded by the CEA insertion curves, as provided in the analysis discussed in Sections 3.3.2 through 3.3.4 of this safety evaluation.

## 4.0 REGULATORY COMMITMENTS

The licensee proposed four regulatory commitments to support the TS changes. Regulatory commitments specify the items for which the licensees volunteer to perform in support of their licensing applications. The regulatory commitments do not require prior NRC approval of subsequent changes, and therefore, they are not enforceable licensing requirements. In a typical review of license applications, the NRC staff does not use the regulatory commitments as a basis in the safety evaluation for approving license amendments. The four regulatory commitments proposed by the licensee were reviewed by the NRC staff and are addressed below:

Regulatory Commitment 1 states that:

> The Cycle 21 CEA drop time surveillance data will be provided to the NRC to confirm the conclusion of no further degradation, within 60 days of surveillance completion.

The NRC staff reviewed the commitment and finds that it supports the intent of TS LCO 3.1.3.4 requiring the proposed CEA drop times bound the data measured from SR 4.1.3.4. The NRC staff did not use this commitment as a partial basis for its findings in this safety evaluation.

Regulatory Commitment 2 states that:

> The Cycle 21 CEA drop time data will be analyzed to validate CEA insertion curve remains within the analysis requirements, within 60 days of surveillance completion.

This commitment was modified and elevated to a proposed license condition in the October 8, 2015, supplement. See Section 3.4 of this safety evaluation for the full discussion of this license condition.

Regulatory Commitment 3 states that:

> The limiting accident events will be evaluated for a CEA holding coil decay time of 0.8 seconds.

The analysis proposed in the commitment was submitted to the NRC in the supplement dated August 14, 2015, to be used as supporting information for the licensee's application, so the NRC staff does not consider this information to be a regulatory commitment, as it does not commit to any future action. The NRC staff review of the analysis is documented in Section 3.3.3.4 of this safety evaluation.

Regulatory Commitment 4 states that:

> The radial power fall-off curve limits shall be verified each cycle as part of the Westinghouse reload analysis methodology until a new licensing basis long term fuel methodology is approved for Waterford 3. Upon NRC approval of a new long term fuel evaluation model and associated methods that explicitly account for thermal conductivity degradation (TCD) that is applicable to Waterford Unit 3 design, Entergy will, within 6 months:
>
> a) Demonstrate that Waterford Unit 3 safety analysis remain conservatively bounded in licensing basis analyses when compared to the NRC-

approved new long term fuel evaluation model that is applicable to Waterford Unit 3 design, and/or

b) Provide a schedule for reanalysis using the NRC-approved new long term fuel evaluation model that is applicable to Waterford 3 design for any affected licensing basis analyses.

The NRC staff reviewed this regulatory commitment and finds that it supports the analysis of TCD for WF3, as discussed in Section 3.3.4 of this safety evaluation. However, the NRC staff did not use this regulatory commitment as a partial basis for their findings in this safety evaluation regarding TCD.

Based on its review, the NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to Regulatory Commitments 1 and 4 could best be provided by the licensee's administrative processes, including its commitment management program.

## 5.0 SUMMARY

The proposed TS LCO 3.1.3.4 would increase the maximum arithmetic average of all CEA drop times from less than or equal to 3.0 seconds to less than or equal to 3.2 seconds and increased the maximum individual CEA drop times from less than or equal to 3.2 seconds to less than or equal to 3.5 seconds. The NRC staff finds that the reanalysis in support of the proposed TS uses the NRC-approved methodology, and that the results of the reanalysis of the FSAR, Chapter 15, events meet the requirements of GDC 10 related to the fuel integrity and GDC 15 related to the reactor coolant pressure boundary limits. Also, the inclusion of the proposed rod drop times in the TS meets the 10 CFR 50.36 requirements specified in Criterion (2) regarding initial condition used in the FSAR Chapter 15 analysis, and Criterion (3) regarding the systems, structures and components credited in the FSAR Chapter 15 analysis for consequences mitigation. Therefore, the NRC staff concludes that the proposed changes to TS LCO 3.1.3.4 are acceptable.

## 6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment on October 26, 2015. The State official had no comments.

## 7.0 ENVIRONMENTAL CONSIDERATION

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 published it in the *Federal Register* on September 8, 2015 (80 FR 53892). 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.

## 8.0 PUBLIC COMMENTS

On September 8, 2015, the NRC staff published a "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed Significant Hazards Consideration Determination, and Opportunity for a Hearing," in the *Federal Register* associated with the proposed amendment request (80 FR 53892). In accordance with the requirements in 10 CFR 50.91, "Notice for public comment: State consultation," the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. One public comment was received regarding the proposed amendment (ADAMS Accession No. ML15292A338). Some of the issues discussed in the public comment do not specifically pertain to the proposed NSHC determination. However, the NRC staff has addressed both the issues within the scope of the proposed NSHC and those that are not within the scope. A summary of the comment and the NRC staff response is provided below.

The comment states, in part:

> Why are you letting Japanese owned Westinghouse (Toshiba) hide the alleged safety analysis-justification under false pretense of proprietary rights? You must release this literally critical information to the public.

NRC Response:

Only the small portion of the safety analysis (see Section 3.3.4 of this safety evaluation) regarding thermal conductivity degradation was withheld from the public under 10 CFR 2.390, "Public inspections, exemptions, request for withholding." This information was deemed proprietary, in part, because it contains an analysis method that allows for a competitive advantage to be maintained (ADAMS Accession No. ML15272A069). A redacted copy of the submitted proprietary information can be viewed at ADAMS Accession No. ML15268A020.

The comment states, in part:

> You should never be using the arithmetic average, which can be all over the place. The Entergy presentation in April gave Individual CEA drop times 3.2 seconds and proposed raising them to 3.5 seconds. This individual CEA drop

> time seems to be lost in this comment period and must be reinstated, at the minimum....
>
> You must have individual maximums, along with the average of all, especially when it is arithmetic average.

NRC Response:

The proposed amendment does contain individual CEA drop time maximums. The proposed individual CEA drop time maximum is less than or equal to 3.5 seconds. If any CEA drop time is over 3.5 seconds during the test, then the reactor does not pass SR 4.1.3.4 and cannot proceed to power operations. The proposed arithmetic drop time average is calculated by using all of the individual CEA drop times.

The comment states, in part:

> So, this is serious business and not to be played around with like you are doing. It's about safely shutting down the nuclear reactor.
>
> The April Entergy presentation stated that any of these problems may be at the root of the slow insertion speed. These need to be addressed:
> "Potential Causes Plant Primary Side Modifications
> Steam Generator replacement
> Reactor Vessel head replacement
> Reactor Vessel head replacement
> CEA replacement
> Transition to Next Generation Fuel Product"

NRC Response:

Since the licensee demonstrated that all acceptance criteria continue to be met when using conservative CEA drop times in their reanalysis of the FSAR, Chapter 15, events, the root causes of the slow insertion speed does not need to be addressed. The NRC staff reviewed the entire submittal and supplements from Entergy and determined that the increased drop times continue to support the safe operation and shutdown ability of WF3. The radiological consequences of a transient or accident are not changed by the proposed rod drop times. Regardless of the root causes of the slow insertion speed, WF3 must continue to operate within the bounds of its license as modified by this amendment. For additional information regarding the possible causes of the slower CEA drop times and the associated regulations, please see Section 3.1.2 of this safety evaluation.

## 9.0 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: S. Sun
R. Beaton
P. Clifford

Date: November 13, 2015

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

*/RA/*

April L. Pulvirenti, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 246 to NPF-38
2. Safety Evaluation (non-proprietary)
3. Safety Evaluation (proprietary)

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**ADAMS Accession No.: ML15289A143** ***concurrence via memo dated**
****NLO w/comments and subject to no hearing request**

| OFFICE | NRR/DORL/LPLIV-2/PM | NRR/DORL/LPLIV-2/LA | NRR/DSS/SRXB/BC* | NRR/DRA/ARCB/BC |
|---|---|---|---|---|
| NAME | MOrenak | PBlechman | CJackson | UShoop |
| DATE | 10/22/2015 | 10/21/2015 | 10/05/2015 | 10/23/2015 |
| OFFICE | NRR/DIRS/STSB/BC | OGC** | NRR/DORL/LPLIV-2/BC | NRR/DORL/LPLIV-2/PM |
| NAME | RElliot | JLindell | MKhanna | APulvirenti |
| DATE | 10/26/2015 | 11/03/2015 | 11/09/2015 | 11/13/2015 |

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