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

February 24, 2014

Mr. George T. Hamrick
Vice President
Brunswick Nuclear Plant
P. O. Box 10429
South Port, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 LICENSE AMENDMENTS REGARDING CHANGES TO TECHNICAL SPECIFICATIONS FOR EXTENSION OF COMPLETION TIME FOR AN INOPERABLE EMERGENCY DIESEL GENERATOR (TAC NOS. ME8893 AND ME8894)

Dear Mr. Hamrick:

The Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment Nos. 264and292 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for Brunswick Steam Electric Plant, Unit Nos. 1 and 2, respectively, in response to your letter dated June 19, 2012, as supplemented by letters dated January 21, May 14, and August 29, 2013, and January 22, 2014. The amendments extend the completion time for an inoperable emergency diesel generator from 7 to 14 days.

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

Sincerely,

Farideh E. Saba

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

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 264 to DPR-71
2. Amendment No. 292 to DPR-62
3. Safety Evaluation

cc w/enclosures: Distribution via ListServ

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 264
License No. DPR-71

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

   A. The application for amendment by Duke Energy Progress, Inc., dated June 19, 2012, as supplemented by letters dated January 21, May 14, and August 29, 2013, and January 22, 2014, 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 Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 264, are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the 2014 Unit 1 refueling outage (B120R1).

FOR THE NUCLEAR REGULATORY COMMISSION

for

Jessie F. Quichocho, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: February 24, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 264

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace page 4 of Renewed Facility Operating License DPR-71 with the attached page 4.

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

Remove	Insert
3.8-3	3.8-3
3.8-4	3.8-4
3.8-5	3.8-5
3.8-6	3.8-6*
3.8-7	3.8-7*

* This TS page is reformatted only.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 264 , are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203 including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 203.

(a) Effective June 30, 1982, the surveillance requirements listed below need not be completed until July 15, 1982. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2 shall apply.

Specification 4.3.3.1, Table 4.3.3-1, Items 5.a and 5.b

(b) Effective July 1, 1982, through July 8, 1982, Action statement "a" of Technical Specification 3.8.1.1 shall read as follows:

ACTION:

a. With either one offsite circuit or one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within two hours and at least once per 12 hours thereafter; restore at least two offsite circuits and four diesel generators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(3) Deleted by Amendment No. 206.

D. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21 are entitled: "Physical Security Plan, Revision 2," and "Safeguards Contingency Plan, Revision 2," submitted by letter

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One offsite circuit inoperable for reasons other than Condition A or B.	C.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	C.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one 4.16 kV emergency bus concurrent with inoperability of redundant required feature(s)
	AND		
	C.3	Restore offsite circuit to OPERABLE status.	72 hours AND 17 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One DG inoperable for reasons other than Condition B.	D.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	D.2	Evaluate availability of supplemental diesel generator (SUPP-DG)	2 hours AND Once per 12 hours thereafter
	AND		
	D.3	Declare required feature (s), supported by the inoperable DG, inoperable when the redundant required feature (s) are inoperable.	4 hours from discovery of Condition D concurrent with inoperability of redundant required feature (s)
	AND		
	D.4.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	D.4.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.5	Restore DG to OPERABLE status.	7 days from discovery of unavailability of SUPP-DG AND 24 hours from discovery of Condition D entry ≥ 6 days concurrent with unavailability of SUPP-DG AND 14 days AND 17 days from discovery of failure to meet LCO 3.8.1.a or b
E. Two or more offsite circuits inoperable for reasons other than Condition B.	E.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	AND		
	E.2	Restore all but one offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One offsite circuit inoperable for reasons other than Condition B. AND One DG inoperable for reasons other than Condition B.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition F is entered with no AC power source to any 4.16 kV emergency bus. ------------------------------ F.1 Restore offsite circuit to OPERABLE status. OR F.2 Restore DG to OPERABLE status.	 12 hours 12 hours
G. Two or more DGs inoperable.	G.1 Restore all but one DG to OPERABLE status.	2 hours
H. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. One or more offsite circuits and two or more DGs inoperable. OR Two or more offsite circuits and one DG inoperable for reasons other than Condition B.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	-----------------------------NOTES----------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. 3. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3750 V and ≤ 4300 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

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

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 292
License No. DPR-62

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

   A. The application for amendment by Duke Energy Progress, Inc., dated June 19, 2012, as supplemented by letters dated January 21, May 14, and August 29, 2013, and January 22, 2014, 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 Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 292 , are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the 2014 Unit 1 refueling outage (B120R1).

FOR THE NUCLEAR REGULATORY COMMISSION

for

Jessie F. Quichocho, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: February 24, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 292

RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace page 3 of Renewed Facility Operating License DPR-62 with the attached page 3.

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

Remove	Insert
3.8-3	3.8-3
3.8-4	3.8-4
3.8-5	3.8-5
3.8-6	3.8-6*
3.8-7	3.8-7*

* This TS page is reformatted only.

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source, and special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2.

(6) Duke Energy Progress, Inc. shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated November 22, 1977, as supplemented April 1979, June 11, 1980, December 30, 1986, December 6, 1989, July 28, 1993, and February 10, 1994 respectively, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 292, are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One offsite circuit inoperable for reasons other than Condition A or B.	C.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	C.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one 4.16 kV emergency bus concurrent with inoperability of redundant required feature(s)
	AND		
	C.3	Restore offsite circuit to OPERABLE status.	72 hours AND 17 days from discovery of failure to meet LCO 3.8.1.a or b

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One DG inoperable for reasons other than Condition B.	D.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	2 hours AND Once per 12 hours thereafter
	AND		
	D.2	Evaluate availability of supplemental diesel generator (SUPP-DG)	2 hours AND Once per 12 hours thereafter
	AND		
	D.3	Declare required feature (s), supported by the inoperable DG, inoperable when the redundant required feature (s) are inoperable.	4 hours from discovery of Condition D concurrent with inoperability of redundant required feature (s)
	AND		
	D.4.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	D.4.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.5	Restore DG to OPERABLE status.	7 days from discovery of unavailability of SUPP-DG AND 24 hours from discovery of Condition D entry ≥ 6 days concurrent with unavailability of SUPP-DG AND 14 days AND 17 days from discovery of failure to meet LCO 3.8.1.a or b
E. Two or more offsite circuits inoperable for reasons other than Condition B.	E.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	AND		
	E.2	Restore all but one offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One offsite circuit inoperable for reasons other than Condition B. AND One DG inoperable for reasons other than Condition B.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition F is entered with no AC power source to any 4.16 kV emergency bus. ------------------------------	
	F.1 Restore offsite circuit to OPERABLE status.	12 hours
	OR	
	F.2 Restore DG to OPERABLE status.	12 hours
G. Two or more DGs inoperable.	G.1 Restore all but one DG to OPERABLE status.	2 hours
H. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.	H.1 Be in MODE 3.	12 hours
	AND	
	H.2 Be in MODE 4.	36 hours
I. One or more offsite circuits and two or more DGs inoperable. OR Two or more offsite circuits and one DG inoperable for reasons other than Condition B.	I.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	---------------------------NOTES-------------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. 3. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3750 V and ≤ 4300 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	 31 days

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 264 AND 292

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, INC.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By application dated June 19, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12173A112), as supplemented by letters dated January 21, 2013 (ADAMS Accession Nos. ML13022A572 and ML13022A573), May14, 2013 (ML13141A405), August 29, 2013 (ADAMS Accession No. ML13260A252), and January 22, 2014 (ML14041A213), Duke Energy Progress, Inc. (the licensee) requested changes to the technical specifications (TSs) for the Brunswick Steam Electric Plant (BSEP). The proposed changes would revise TS 3.8.1 Required Action D.4 to extend the completion time (CT) associated with an inoperable emergency diesel generator (EDG) from 7 days to 14 days. The change is requested to allow sufficient time to perform adequate preventative maintenance to ensure EDG reliability and availability. The proposed change also permits avoidance of unplanned plant shutdown for EDG corrective maintenance. In addition, the licensee proposed a supplemental alternating current (AC) power source, SUPP-DG, as a defense-in-depth measure to be consistent with NUREG-0800, Branch Technical Position (BTP) 8-8.

2.0 REGULATORY EVALUATION

The following U. S. Nuclear Regulatory Commission (NRC) requirements and guidance are applicable to the NRC staff's review of the licensee's amendment request (LAR):

General Design Criterion (GDC) 17, "Electric power systems," of Appendix A, to Title 10, of the *Code of Federal Regulations* (CFR), Part 50, requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of

Enclosure

their simultaneous failure under operating and postulated accident and environmental conditions.

GDC 18, "Inspection and testing of electric power systems," of Appendix A, to 10 CFR, Part 50, requires, in part, that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards to assess the continuity of the systems and the condition of their components.

In the LAR, Enclosure 1, Section 5.2, the licensee stated that the BSEP design was reviewed for construction under the "General Design Criteria for Nuclear Power Plant Construction" issued for comment by the Atomic Energy Commission in July 1967 and is committed to meet the intent of the GDC, published in the *Federal Register* on May 21, 1971, as Appendix A to 10 CFR Part 50.

Title 10 CFR 50.36, "Technical Specifications," requires, in part, that the TS shall be included by applicants for a license authorizing operation of a production or utilization facility. 10 CFR 50.36(c) requires that TS include items in five specific categories related to station operation. These categories are (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls. The proposed change to the BSEP TS relates to the LCO category.

Title 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires, in part, that performing maintenance activities shall not reduce the overall availability of the systems, structures and components, which are important to safety of the plant.

Title 10 CFR 50.63, "Loss of all alternating current power," requires, in part, that a nuclear power plant shall be able to withstand for a specified duration, and recover from a complete loss of offsite and onsite AC sources (i.e., a station blackout (SBO)).

Regulatory Guide (RG) 1.93, "Availability of Electric Power Sources," provides guidance with respect to operating restrictions or CT if the number of available AC sources is less than that required by the TS LCO. In particular, this guide recommends a maximum CT of 72 hours for an inoperable onsite or offsite AC source.

RG 1.155, "Station Blackout," provides guidance for complying with the 10 CFR 50.63 that requires nuclear power plants to be capable of coping with an SBO event for a specified duration.

NUREG-0800, BTP 8-8, "Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions," dated February 2012 (ADAMS Accession No. ML113640138) provides guidance to the NRC staff in reviewing LARs for licensees proposing a one-time or permanent TS change to extend an EDG Allowed Outage Time beyond 72 hours.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in allowed outage times. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensees evaluation of the risk associated with a proposed CT TS change, as discussed below.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" describes an acceptable approach for determining whether the quality of the Probabilistic Risk Assessment (PRA), in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors.

Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 provides general guidance for evaluating the technical basis for proposed risk-informed changes. Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," provides guidance on evaluating PRA technical adequacy. More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," which includes CT changes as part of risk-informed decisionmaking. Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes increase core damage frequency (CDF) or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

3.1 Description of the Brunswick Units 1 and 2 AC Power System

Normal plant power for each unit's auxiliaries is supplied by the 24 kilovolt (kV) to 4.16 kV unit auxiliary transformer (UAT), which is connected to its associated generator output. The startup power for each unit's auxiliaries is supplied by the 230 kV to 4.16 kV startup transformer that is fed from the 230 kV switchyard. The startup transformers also provide power required in the event of a design-basis accident condition. Standby power is provided by four EDGs that are started automatically on loss of voltage on the 4.16 kV buses or on a loss-of-coolant accident (LOCA) signal.

The auxiliary power to major loads is supplied at 4160 voltage. The nonsafety-related loads are fed from 4160 volt (V) Buses 1B, 1C, 1D, and Common A for Unit 1; and Buses 2B, 2C, 2D, and Common B for Unit 2. The safety-related 4160 V system buses for Units 1 and 2 are E1, E2, E3, and E4. The safety-related buses E1 and E2 are normally fed from upstream buses 1D and 1C respectively. Similarly, the safety-related buses E3 and E4 are normally fed from upstream buses 2D and 2C respectively. The upstream buses (B, C, D, and the common) are in turn fed from the respective unit's 24 kV main generator through a UAT during normal plant operation, and from the 230 kV offsite power through the respective startup auxiliary transformer (SAT) during startup and shutdown operation. Buses 1C, 1D, 2C, and 2D are provided with an automatically initiated, automatically executed, quick dead bus transfer. The scheme is capable of quickly transferring each bus section and its loads from the normal source (i.e., UAT) to the preferred source (i.e., SAT) in the event of loss of the normal source or unit trip. In the case of loss of offsite power, each safety-related bus (E1 through E4) is fed from its respective EDG. All AC loads necessary to the safe shutdown of the plant under accident or nonaccident conditions are fed from this emergency buses (E-buses) distribution system.

The four 4.16 kV E-buses, E1 through E4, can be connected to each other through two normally open tie-breakers. The electrical design, breaker control logic, and procedures prevent any two 4.16 kV redundant E-buses from being tied together except during an SBO or Appendix R fire event.

Four EDGs provide standby power for the engineered safety features on the loss of the normal power sources. Once an EDG is automatically connected to the E-bus, the logic recognizes which unit is in an accident condition and automatically starts the appropriate engineered safety features, according to a prescribed timed sequence.

Each EDG has a continuous rating of 3500 kilo-Watt (kW) and a 2000-hour rating of 3850 kW adequately sized for the following conditions:

- Design basis accident on one unit and orderly shutdown of the other unit under loss of offsite power (LOOP) conditions with three EDGs operating (i.e., LOCA/LOOP loading).
- Simultaneous safe shutdown of both units under LOOP conditions with three EDGs operating (i.e., LOOP loading).
- SBO loading with one EDG available (safe shutdown of the nonblackout unit while supplying the SBO coping loads of the blackout unit).

In the LAR, Enclosure 1, Section 3.3, the licensee stated that BSEP utilized the alternate alternating current (AAC) source operation approach for compliance with the requirement of 10 CFR 50.63. BSEP is subject to a minimum station blackout coping capability of four hours with the EDG reliability target of 0.975. A SBO is assumed to occur in only in one of the units. The non-blackout unit (after assuming a single EDG failure) is capable of sharing a single EDG with the blackout unit. With only one EDG available, it is necessary to load strip and cross-tie the 4.16 kV and 480 V buses to provide power to both units. Within first hour of SBO, the crosstie is established between the 4.16 kV buses of SBO unit and non-SBO unit. After the crosstie is established, the AAC power source can support the coping requirements (i.e., supply battery chargers) in the blackout unit.

In addition, as a defense-in-depth measure, two 200 kW, 480 V alternating current (AC) 60 Hertz diesel generators (DGs), referenced as severe accident management alternative DGs, are available to supply the blackout unit battery chargers if AC power cannot be restored to any E-bus of the blackout unit (i.e., if the crosstie is not possible).

3.2 <u>Proposed TS Changes</u>

According to TSs, the BSEP is subject to a dual unit shutdown should an EDG outage occur, with the EDG not restored to operable status within 7 days. In the LAR, the licensee stated that the purpose of the proposed change is to extend the TS CT for an inoperable EDG from 7 days to 14 days. The 14-day CT is needed to (1) provide the necessary time to support planned EDG voltage regulator, governor, and starting air replacements, and (2) reduce the likelihood and unnecessary burden of a dual unit shutdown should an unplanned EDG outage occur with the units at power by providing additional time to repair and reestablish operability of the inoperable EDG. To justify the 7-day CT extension, the licensee will install a supplemental AC power source (i.e., a supplemental DG) with the capability to power any of the four 4.16 kV E-buses within 1 hour from the SBO event, and with the capacity to bring the affected unit to cold shutdown, and be consistent with BTP 8-8.

The licensee stated that any EDG maintenance activities if combined with the planned EDG component replacement activities over the proposed CT extension would also help in reducing the number of entries into the TS Actions.

The proposed TS 3.8.1 changes commensurate with the proposed 14-days CT for an inoperable EDG are as follows:

TS 3.8.1 - Required Action and Completion Time – New Required Action

> A new Required Action D.2 to "Evaluate availability of supplemental diesel generator (SUPP-DG)" with a CT of "2 hours AND Once per 12 hours thereafter."

TS 3.8.1 - Completion Time - Required Action D.4

> The CT for Required Action D.4 (revised to D.5 due to new Required Action D.2) is proposed to be extended from 7 days to 14 days provided the SUPP-DG is available. The existing 7-day CT will be maintained if the SUPP-DG is unavailable. Additionally, a CT is proposed to limit the time the SUPP-DG is unavailable during the extended CT to 24 hours.

TS 3.8.1 - Maximum Completion Time - Required Actions C.3 and D.4

> The maximum CT for Required Actions C.3 and D.4 (D.4 revised to D.5 due to new Required Action D.2) are proposed to be extended from currently 10 days to 17 days. The maximum CT limits the total time that LCO 3.8.1 is not met while simultaneously in Conditions C and D. This maximum CT is the sum of CT for Required Actions C.3 and D.4. Due to the proposed increase in the CT time for D.4 from 7 to 14 days, the commensurate change to the maximum CT for Required Actions C.3 and D.4 is an increase of 7 days (from 10 days to 17 days).

3.3 SUPP-DG Details

The licensee proposed a supplemental AC power source, SUPP-DG, as a defense-in-depth measure to be consistent with BTP 8-8. The licensee provided following details of the SUPP-DG.

The proposed SUPP-DG would be able to provide power to any one of the four E-buses. The SUPP-DG will have a continuous rating of 4000 kW continuous (for 6850 hours/year) and short-time rating of 4300 kW (for less than 2 hours out of any 24-hour period). The 4000 kW rating of the SUPP-DG exceeds that of a station EDG continuous rating of 3500 kW; thus, the SUPP-DG can substitute for any one of the four EDGs under SBO conditions, and has the capacity to bring the affected unit to cold shutdown, if needed (i.e., if the offsite power is not recovered in a timely manner).

According to the diagram provided in the LAR, Enclosure 1, the proposed SUPP-DG can be tied to one of the plant balance-of-plant (BOP) Buses located on either Unit 1 (i.e., 1C, 1D), or Unit 2 (i.e., 2C, 2D) which in turn supply power to the connected Class 1E bus. There will be no direct interface between the SUPP-DG and the plant EDGs. The SUPP-DG will be connected to one of the BOP buses during the SUPP-DG load testing. The SUPP-DG will be connected such that it can supply power only to one 4.16 kV E-Bus at a time in accordance with the procedures. In the letter dated January 21, 2013, the licensee provided details of the protective relaying at the SUPP-DG output breaker. The new BOP bus breakers will be normally open to provide redundant means of separation between the BOP buses. The logic of the new BOP bus breakers will be controlled such that only one of the four breakers will be selected and closed at any time. The BOP bus breaker to be used for connection of SUPP-DG will be selected and closed from the SUPP-DG engine control panel.

The licensee stated that factory testing will be performed by the manufacturer on the assembled SUPP-DG. Factory testing will include 24-hour load capability, largest single load start, short-time rating, largest single load reject, start and load acceptance tests. Site acceptance testing will include testing prior to connection to the plant electrical system and load testing to the 4.16 kV BOP Buses. Upon implementation of the modification, the SUPP-DG will be subject to monthly load testing to a 4.16 kV BOP Bus, to ensure all its auxiliary support systems are available or operational.

The SUPP-DG will be of commercial-grade type, non Class 1E, permanently-installed inside the plant protected area, and outside the existing power block building; east of the switchyard and north of the transformer yard. In the letter dated January 21, 2013, the licensee provided drawings showing the proposed physical location of major components of the SUPP-DG, such as diesel enclosure, electrical enclosure, mechanical enclosure, radiator, fuel storage tank, refill station for the tank. The SUPP-DG and associated mechanical and electrical equipment will be primarily housed in three outdoor weather enclosures (i.e., Diesel Enclosure, Mechanical Enclosure, and Electrical Enclosure) and mounted on an elevated foundation to protect the SUPP-DG system from flood and storm surge. The 4.16 kV cable from the SUPP-DG to the Turbine Building is also flood-resistant. The closed loop radiator cooling system of SUPP-DG will also be mounted on the elevated foundation. The outdoor weather enclosures and radiator assembly will be able to withstand wind speed up to 155 mph (3-second gust). The Diesel Enclosure will house the diesel engine and generator, and its support systems and accessories including Lube Oil, Air Intake, Cooling Water, Fuel Oil, Redundant Dual Turbine Motor Air Start Systems, and Engine Panel with Engine Instrumentation. The Mechanical Enclosure will house required air start accessories, including air receivers, air compressor and dryer. The Electrical Enclosure will house the SUPP-DG output breaker switchgear, engine/generator control panel, 125 V DC [direct current] battery system with battery charger, 4160:480 V auxiliary power transformer, 480 V automatic transfer switch, and 480 V motor control center. SUPP-DG output breaker and the feeder breakers to the 4.16 kV BOP Buses will be operated from the Electrical Enclosure.

The SUPP-DG system will normally be in a standby configuration and disconnected from the plant 4.16 kV electrical systems. When the plant is not subject to LOOP/SBO, the SUPP-DG house loads (e.g., battery charger, air compressor, heaters) will be powered from a plant 480 V supply through a transfer switch located at the SUPP-DG. Upon LOOP, the SUPP-DG will be capable of starting from its 125 V DC battery system, and then power its house loads once running through its 4160:480 V auxiliary power transformer.

The SUPP-DG will have a 10,000 gallon fuel storage tank mounted on the SUPP-DG foundation. The fuel storage tank will provide a usable supply sufficient for 24 hours of running time at rated continuous load. The required action prior to entering the extended CT will include verifying SUPP-DG availability, which includes fuel tank level greater than or equal to a 24-hour supply. The fuel storage tank can be replenished via a refill station for the tank. In the letter dated January 21, 2013, the licensee stated that the refill station will be located at grade elevation adjacent to the elevated platform upon which the 10,000 gallon fuel oil storage tank will be mounted. The fuel oil storage tank will be refueled from grade elevation via fuel tanker truck.

In the LAR, Enclosure 1, Section 3.4.4, the licensee also stated that the SUPP-DG will be equipped with fire detection and suppression. The Diesel, Mechanical, and Electrical Enclosures will each be equipped with a fire suppression system for the hazards present. The fire detection will be connected to the plant fire detection system. The SUPP-DG will be monitored for fire hazards during Operator Rounds. Operator rounds will also ensure the availability of fuel oil and the functionality of the SUPP-DG starting batteries (i.e., charging status) and starting air system, and provide an additional level of the SUPP-DG overall monitoring.

SUPP-DG availability will include verification that the load test is performed within 30 days of entry into the extended CT. The proposed TS changes require evaluation of SUPP-DG availability within 2 hours of entry into TS 3.8.1, Condition D, for an inoperable DG. This verification includes an administrative verification of this prior testing. Following initial verification of the SUPP-DG availability, the proposed TS changes provide verification of continued availability on a once per 12 hours frequency.

3.4 Deterministic Evaluation

The staff reviewed the information provided by the licensee in the LAR and supplemental information provided in the letter dated January 13, 2013. In particular, the NRC staff followed the guidance provided in NUREG-0800, BTP 8-8, and RG 1.177, Section 2.2 (traditional engineering consideration) for deterministic evaluation.

The staff finds that the proposed LAR generally meets the requirements as stated in BTP 8-8, and RG 1.177, Section 2.2, as discussed below:

- The proposed SUPP-DG as supplemental power source is of adequate capacity to provide cold shutdown as recommended in BTP 8-8, which states (on page 8-8-3) that the supplemental source must have capacity to bring a unit to safe shutdown (cold shutdown) in case of a LOOP concurrent with a single failure during plant operation. The SUPP-DG meets this requirement.
- The proposed TS changes for checking the availability of SUPP-DG (which includes evaluation of the SUPP-DG availability within 2 hours of entry into TS 3.8.1, Condition D, and once per 12 hours frequency during the CT extension) are adequate and meet the intent of BTP 8-8, which states (on page 8-8-5) that the availability of AAC or supplemental power source shall be checked every 8-12 hours (once per shift).

- The regulatory commitments proposed in the LAR, Enclosure 9, as compensatory measures meet the intent of defense-in-depth measures discussed in BTP 8-8, and Section 2.2.1 of RG 1.177.
- The proposed TS changes will continue to meet the principle that safety-margins are maintained as discussed in Section 2.2.2 of RG 1.177, which states that the sufficient safety margins are maintained if the proposed CT change is not in conflict with the approved codes and standards relevant to the subject; and the proposed CT change does not adversely affect any assumptions or input to the safety analysis. In the LAR, the licensee stated that the proposed TS changes do not alter the plant design nor does it change the assumptions in the safety analysis. The staff finds that LAR meets the requirements of Section 2.2.2 of RG 1.177.

The NRC staff reviewed the proposed SUPP-DG information with respect to size, installation, auxiliaries, and connection to the existing electrical system. Based on its review, the staff finds that the SUPP-DG would be capable of performing its required function, if necessary, during the period of extended CT of an EDG, as recommended in BTP 8-8.

In a supplemental letter dated January 21, 2013, the licensee also stated that to meet the intent of BTP 8-8, in day-to-day operations, Power System Operations (responsible for the operation of the transmission grid) shall contact BSEP Operations-Control Room each business day to discuss the status of the plant and the transmission system and review upcoming plans and work activities for the day. When an EDG becomes inoperable, the plant Integrated Scheduling will require the remaining offsite sources and EDGs to be protected. These controls apply to maintenance and testing activities that place the equipment outside its normal, as designed, configuration.

In this supplemental letter, the licensee further confirmed that a unit can remain in safe shutdown condition without any AC power for the first hour of an SBO event. The EDG reliability target of 0.975 used in the SBO analysis will not be altered. The 14-day CT will not impact the licensee's ability to continue meeting the Maintenance Rule requirements, and the reactor oversight process performance indicator criteria for EDG availability/reliability.

The NRC staff also reviewed whether the proposed TS changes will have any impact on the licensee's compliance with GDC 17, GDC 18, 10 CFR 50.36, 10 CFR 50.65, and 10 CFR 50.63. The staff did not find any adverse impact on continued compliance with these regulatory requirements.

3.4.1 Regulatory Commitments

In LAR, Enclosure 9, the licensee provided the following regulatory commitments, as compensatory measures, prior to implementing the proposed TS 3.8.1 EDG CT Extension:

1. The SUPP-DG will be protected, as defense-in-depth, during the extended DG CT.

2. The SUPP-DG will be routinely monitored during Operator Rounds, with monitoring criteria identified in the Operator Rounds. The SUPP-DG will be monitored for fire hazards during Operator Rounds.

3. Component testing or maintenance of safety systems and important nonsafety equipment in the offsite power systems which can increase the likelihood of a plant transient (i.e., unit trip) or LOOP, will be avoided during the extended DG CT.

4. No discretionary switchyard maintenance will be allowed during the extended DG CT.

5. Weather conditions will be evaluated prior to intentionally entering the extended DG CT and will not be entered if official weather forecasts are predicting severe weather conditions (i.e., tornado or hurricane warnings). Operators will monitor weather forecasts each shift during the extended DG CT. If severe weather or grid instability is expected after a DG outage begins, station managers will assess the conditions and determine the best course for returning the DG to an operable status.

6. Licensed Operators and Auxiliary Operators, for the operating crews on-shift when the extended DG CT is in use, will be briefed on the DG work plan, the revised TS 3.8.1, and procedural actions regarding LOOP, SBO, and SUPP-DG alignment and use prior to entering the extended DG CT.

7. Licensed Operators and Auxiliary Operators will be appropriately trained on the purpose and use of the SUPP-DG and the revised AOP actions. Personnel performing maintenance on the SUPP-DG will be appropriately trained.

8. The High Pressure Coolant Injection (HPCI) pump, the Reactor Core Isolation Cooling (RCIC) pump, and the Residual Heat Removal (RHR) pump associated with the operable DG will not be removed from service for elective maintenance activities during the extended DG CT.

The NRC staff finds the above Regulatory Commitments, as compensatory measures, acceptable.

3.4.2 Deterministic Evaluation Conclusion

Based on the review discussed above, the NRC staff concludes that the licensee's request to revise TS 3.8.1 to extend the CT of an inoperable EDG to operable status from the current 7 days to 14 days is acceptable from a deterministic perspective. The staff finds the proposed TS changes will have no adverse impact on the licensee's compliance with 10 CFR 50, Appendix A, GDC 17 and GDC 18, 10 CFR 50.36, 10 CFR 50.65, and 10 CFR 50.63.

3.5 <u>Risk-Informed Evaluation</u>

The key information used in the NRC staff's risk-informed evaluation is contained in the licensee's application dated June 19, 2012 and supplemented by letter dated January 21, 2013. Using the guidance in SRP Sections 19.2 and 16.1, the NRC staff reviewed the proposed amendment using the three-tiered approach discussed in RG 1.177 and the five key principles of risk-informed decisionmaking discussed in RG 1.174 and RG 1.177 as summarized below. Key Principle 4 is evaluated below in the SE Section 3.5.2.

3.5.1 Key Principles 1, 2, 3, and 5

The traditional engineering evaluation addresses key principles 1, 2, 3, and 5 of the NRC integrated approach to risk-informed decisionmaking: compliance with current regulations, defense-in-depth, safety margins, and performance monitoring strategies.

Key Principle 1 – Compliance with Current Regulations

The staff finds that the proposed LAR generally meets the requirements as stated in BTP 8-8, and RG 1.177, Section 2.2. The staff also reviewed whether the proposed TS changes will have any impact on the licensee's compliance with GDC 17, GDC 18, 10 CFR 50.36, 10 CFR 50.65, and 10 CFR 50.63. The staff did not find any adverse impact on continued compliance with these regulatory requirements. See Section 3.4 for detailed discussions.

Key Principle 2 – Defense-in-Depth

The licensee proposed a supplemental AC power source, SUPP-DG, as a defense-in-depth measure to be consistent with BTP 8-8. See Section 3.3 for detailed discussions.

Key Principle 3 – Safety Margins

The staff finds that the proposed TS changes will continue to meet the principle that safety-margins are maintained as discussed in Section 2.2.2 of RG 1.177.

Key Principle 5 – Performance Measurement Strategies

In response to the RAI, the licensee further confirmed in letter dated January 21, 2013, that a unit can remain in safe shutdown condition without any AC power for the first hour of an SBO event. The EDG reliability target of 0.975 used in the SBO analysis will not be altered. The 14-day CT will not impact the licensee's ability to continue meeting the Maintenance Rule requirements, and the reactor oversight process performance indicator criteria for EDG availability/reliability.

3.5.2 Key Principle 4 – Risk is Consistent with Commission's Safety Goal Policy Statement

The evaluation presented below addresses key principle 4 of the NRC staff's risk-informed decisionmaking process described in RG 1.174. Specifically, when a proposed change results in an increase in risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (51 FR 30028).

3.5.2.1 <u>Tier 1: PRA Capability and Insights</u>

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

PRA Quality

The objective of the PRA quality review was to determine whether the BSEP PRA used in evaluating the proposed change is of sufficient scope, level of detail, and technical adequacy for this application. The NRC staff review evaluated the PRA quality information provided by the licensee in their submittal, including industry peer review results and self-assessments performed by the licensee.

Internal Events Model (includes internal flooding)

The BSEP PRA model addresses both CDF and large early release frequency (LERF) for internal events at full power. The model has been updated to reflect power uprate conditions. The licensee has processes for configuration control of the BSEP PRA model to reflect plant modifications and procedural changes. No outstanding changes, not yet incorporated into the PRA model or dispositioned as not relevant to the model were identified by the licensee that would have a significant impact on the model.

The BSEP internal events PRA model (including internal flooding) was subjected to an industry peer review in 2010, using the American Society of Mechanical Engineers (ASME) PRA Standard ASME/ANS [American Nuclear Society] RA-Sa-2009, as endorsed by the NRC in RG 1.200, Rev 2. The licensee resolved many of the findings from this review through additional analysis. There were six findings from this review where the internal events model was found not to conform to capability category II of the standard for certain supporting requirements (SRs). One of these was a documentation issue with no impact on this application. The remaining five findings were dispositioned by the licensee; NRC staff review of the impact on this application is discussed below:

DA-C8: The peer review team found that plant-specific data concerning EDG standby time was not collected or used by the PRA model. The licensee has provided additional information demonstrating that plant-specific data was used. Based on this information, the finding has been sufficiently resolved with respect to this application.

LE-C3: The licensee did not review LERF sequences to determine whether repair of equipment could be credited. The licensee's approach was conservative because it did not credit equipment repair. Therefore, this finding is judged to have negligible impact on the conclusions of the application.

LE-C10, LE-C12: The licensee did not review LERF sequences to determine whether engineering analyses can support continued equipment during an accident sequence (including postcontainment failure) in a manner that could reduce LERF. The licensee did not credit equipment survivability, which is a conservative approach. Therefore, this finding is judged to have negligible impact on the conclusions of the application.

LE-C13: The licensee performed a conservative, rather than realistic containment bypass analysis. This approach would tend to overestimate LERF and therefore would not impact the conclusions of the application.

Based on consideration of the gaps to capability category II of the PRA standard and their disposition for this application, the NRC staff finds that the quality of the BSEP internal events PRA is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

Internal Fires Model

Internal fires were addressed by fire PRA techniques described by NUREG/CR-6850. The licensee described the approach taken for the fire PRA as a scenario-by-scenario analysis of unscreened compartments accounting for the relative location of ignition sources and targets. Fire damage calculations were performed to determine the extent of potential damage from each postulated fire source.

The BSEP fire PRA was subjected to an industry peer review in 2011, using the ASME PRA Standard ASME/ANS RA-Sa-2009, as endorsed by the NRC in RG 1.200, Rev 2. This review utilized the process described by NEI-07-12. The licensee resolved many of the findings from this review through additional analysis. There were 28 findings from this review where the fire PRA model was found not to conform to capability category II of the standard for certain SRs. Twelve findings were related to documentation and were determined not to impact this application. The remaining 16 findings were dispositioned by the licensee; NRC staff review of the impact on this application is discussed below:

SC-B2: Changes to success criteria for fire PRA relied on engineering judgment rather than analytical methods. The licensee provided additional information describing new engineering calculations, thermal hydraulic analyses, and simulator runs to confirm any success criteria that were previously established using engineering judgment. Based on this response indicating that missing evaluations have been completed, the NRC staff concludes that the finding has been sufficiently resolved with respect to this CT extension request.

QU-D5, FQ-E1, and FQ-F1: The licensee's initial submittal did not provide an adequate description of their cutset review. Subsequently, the licensee performed an expanded cutset review that included a larger scope and improved documentation. Based on this supplemental response indicating that the missing reviews have been completed, the NRC staff concludes that the finding has been sufficiently resolved with respect to this CT extension request.

QU-E3, FSS-E3, FSS-H5, UNC-A1, LE-F3, LE-G2, LE-G4: The licensee's uncertainty quantification was limited in scope in that it did not include statistical representation of fire PRA parameters; application of the state of knowledge correlation; or a LERF evaluation. Consistent with RG 1.177, the NRC staff determined that uncertainties associated with the proposed CT change would tend to similarly affect the base case and the changed case and would therefore not appreciably impact comparison to the acceptance guidelines. Therefore, these findings were judged to have negligible impact on the conclusions of the application.

FSS-C1: A heat release rate (HRR) associated with motor fires (69 kW) was used for pump electrical fires rather than the pump electrical HRR of 211 kW that is recommended by NUREG/CR-6850, Table G-1.

The licensee performed a sensitivity study and determined that the impact of using the larger zone of influence associated with the 211 kW HRR was negligible. Based on the results of the sensitivity study, the NRC staff concludes that the finding has been sufficiently resolved with respect to this CT extension request.

FSS-C4: Generic severity factors of 1.0 were applied to most fire scenarios. Assigning a severity factor of 1.0 in lieu of plant-specific data was determined to be conservative and would therefore not impact the conclusions of this application.

FSS-D9: Specific targets that are susceptible to smoke damage were not identified and incorporated into fire scenarios. The licensee provided additional information to address this finding, including a qualitative assessment of spoke effects on equipment relevant to this application. Based on this response indicating that the missing evaluation has been completed, the NRC staff concludes that the finding has been sufficiently resolved with respect to this CT extension request.

CF-A1: A failure probability of 1.0 is assigned to circuits where specific conditional failure probabilities have not yet been developed. This is a conservative assumption that would not have an appreciable impact on this application.

Based on consideration of the gaps to capability category II of the ASME PRA standard and their disposition for this application, the NRC staff finds that the quality of the BSEP Fire PRA is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

High Winds and External Flooding Model

The licensee assessed the risk from high winds and external flooding using a PRA model that was subjected to an industry peer review in 2012, using the ASME PRA Standard ASME/ANS RA-Sa-2009, as endorsed by the NRC in RG 1.200, Rev 2. The licensee resolved many of their industry peer review findings through additional analysis. There were four findings from this peer review where the high winds and external flooding PRA model was found not to conform to capability category II of the standard for certain SRs. Three findings were related to documentation and were determined not to impact this application. One finding WPR-A3 identified included some of the structures, systems, or components (SSCs) that were not analyzed for fragility and was later dispositioned by the licensee. These SSCs were treated in a conservative manner (e.g., assumed to always fail during a high-winds event) and therefore this issue is not expected to impact the amendment request.

Based on consideration of the gaps to capability category II of the ASME PRA standard and their disposition for this application, the NRC staff finds that the quality of the BSEP High Winds and External Flooding PRA is sufficient to support the proposed license amendment.

Seismic

The licensee assessed the risk from seismic events using a bounding quantitative analysis that evaluated the contribution to CDF and LERF from the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE). This analysis used the BSEP internal events PRA model and conservatively assumed a conditional failure probability of 1.0 for all nonsafety-related equipment and all equipment not seismically qualified to the OBE or SSE. The assumed plant response to a seismic event was a nonrecoverable LOOP. The licensee reported that CDF and LERF values for the proposed case (i.e., with the CT extension) were comparable to the base case (i.e., no extended CT applied) and therefore the NRC staff concluded that seismic events are not significant to the regulatory decision.

Other Hazards

The licensee provided a qualitative evaluation of other hazards including nearby facility accidents, aircraft impact, industrial accidents, military accidents, pipeline accidents, hydrogen storage failures, and transportation accidents. This evaluation considered Individual Plant Examination of External Events results and whether they are still applicable to the as-built and operated facility and to assess any impact on the EDG CT evaluation. The NRC staff reviewed this evaluation and concluded that the licensee adequately demonstrated that these hazards are not significant to the regulatory decision and therefore a quantitative PRA evaluation is not required.

PRA Results and Insights

The licensee's baseline PRA model was modified in several ways to assess the change in risk associated with the proposed CT extension. First, a new test and maintenance unavailability event was added to represent an additional 14 days of EDG unavailability per year. Second, logic related to the supplemental DG and associated support equipment (e.g., output breakers) was added. The staff noted that the supplemental DG is only credited as backing up an EDG that is in extended maintenance, not during any other time during plant operations.

The licensee presented the following risk results:

Risk Metric	Total from all hazards		Acceptance Guideline
	Unit 1	Unit 2	
ICCDP *	4.3 E-07	4.1 E-07	< 1 E-06 (RG 1.177, Rev 1)
ICLERP **	2.0 E-09	3.5 E-09	< 1 E-07 (RG 1.177, Rev 1)
ΔCDF	8.5 E-07 yr^{-1}	8.3 E-07 yr^{-1}	< 1 E-06 yr^{-1} (RG 1.174, Rev 2)
ΔLERF	4.8 E-09 yr^{-1}	6.2 E-09 yr^{-1}	< 1 E-07 yr^{-1} (RG 1.174, Rev 2)

* Incremental Conditional Core Damage Probability
** Incremental Conditional Large Early Release Probability

The calculated risk metrics demonstrate the risk acceptance guidelines of RG 1.174 (for very small changes in CDF and LERF) and RG 1.177 (for a small quantitative impact on plant risk) are met.

Uncertainty and Sensitivity Analysis

The licensee followed elements of the process outlined in NUREG-1855 to identify, screen, and evaluate uncertainties in the PRA models. The sources identified included parametric uncertainty, modeling uncertainty and completeness uncertainty. The licensee presented several qualitative arguments for not performing a full parametric uncertainty on the modified PRA model (i.e., the model reflecting the proposed change). The staff did not complete a review of these arguments but noted instead that RG 1.177 contains guidance on uncertainties with respect to CT extensions. Specifically, RG 1.177 states that uncertainties associated with CT changes tend to similarly affect the base case (i.e., before the change) and the changed case (i.e., after the change) and therefore quantification of uncertainty is unlikely to impact comparison to the risk acceptance guidelines.

The licensee evaluated three categories of model uncertainty: BSEP plant-specific, boiling-water reactor generic, and application-specific. This evaluation identified uncertainty with respect to the models used to quantify certain human error probabilities, LOOP frequencies, and fire ignition frequencies. The licensee performed sensitivity studies to assess the impact of these uncertainties.

Completeness uncertainty was evaluated using the method delineated by NUREG-1855. The staff determined that this assessment was adequate because the risk from each significant hazard group was either a) assessed using a PRA model performed in accordance with a consensus standard or b) shown to be insignificant or c) shown to have a minor impact on the decision.

The NRC staff evaluated the scope and results of the uncertainty and sensitivity analyses and determined that they provide reasonable confidence that the increase in plant risk is consistent with the acceptance guidelines delineated by RG 1.174 and RG 1.177.

Shutdown Risk

The licensee conservatively does not credit the avoidance of additional risk during plant shutdowns on assumption that significant EDG maintenance would now be performed during plant operations rather than during outage periods.

3.5.2.2 *Tier 2:* Avoidance of Risk-Significant Plant Configuration

The licensee identified that concurrent EDG outages should be avoided during any EDG extended CT. This is specifically controlled by TS 3.8.1, which provides a CT of 2 hours when two EDGs are inoperable. Furthermore, the proposed TS changes require evaluation of SUPP-DG availability within 2 hours of entry into TS 3.8.1, Condition D for an inoperable EDG. Following initial verification of the SUPP-DG availability, the proposed TS will require verification once every 12 hours.

The licensee also identified other equipment restrictions to be administratively controlled related to the availability of the HPCI pump, RCIC pump, and RHR pump associated with the operable EDG. Furthermore, the licensee identified restrictions on component testing or maintenance on equipment in the offsite power system that can increase the likelihood of a plant transient during the extended EDG CT. These restrictions include an evaluation of weather conditions and the likelihood of severe weather or grid instability. Based on these evaluations and the subsequent identification of compensatory measures, the NRC staff concludes that the required evaluation was performed and that the licensee's approach is therefore consistent with the guidance in RG 1.177.

3.5.2.3 Tier 3: Risk-Informed Configuration Management

The licensee described its Configuration Risk Management Program (CRMP) which ensures that the risk impact of equipment out of service is properly evaluated prior to performing any work activity. The program provides for proceduralized risk-informed assessment of equipment unavailability using a blended approach of defense-in-depth and PRA insights and requires assessment for both planned and unplanned activities, including emergent conditions resulting in configurations not previously assessed. The CRMP therefore provides an acceptable process per RG 1.177 for evaluation of configuration risk during an extended EDG outage to satisfy the Tier 3 requirements.

3.5.3 Risk-Informed Evaluation Conclusion

The risk impact of the proposed 14-day CT for EDGs as reflected in ΔCDF, ΔLERF, ICCDP, and ICLERP is consistent with the acceptance guidelines specified in RG 1.174, RG 1.177, and staff guidance outlined in Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," of NUREG-0800. The Tier 2 evaluation identified risk-significant plant equipment outage configurations controlled by TSs during any extended EDG outage. The licensee's Tier 3 CRMP satisfies the CRMP requirements of RG 1.177. Therefore, the NRC staff finds that the risk analysis methodology and approach used by the licensee to estimate the risk impact and to manage configuration risk during an extended EDG outage are reasonable and of sufficient quality.

Based on the above, the NRC staff finds the proposed changes to extend the EDG CT from 7 days to 14 days to be acceptable.

4.0 REGULATORY COMMITMENTS

The licensee in Enclosure 9 of its letter dated June 19, 2012, listed the regulatory commitments with a completion schedule prior to implementing the proposed TS 3.8.1 EDG CT Extension. Additional detail is provided in Section 3.4.1 of this SE.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 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 there has been no public comment on such finding (77 FR 63346). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The NRC 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 NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Christopher J. Fong
Vijay K. Goel

Date: February 24, 2014

February 24, 2014

Mr. George T. Hamrick
Vice President
Brunswick Nuclear Plant
P. O. Box 10429
South Port, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 LICENSE AMENDMENTS REGARDING CHANGES TO TECHNICAL SPECIFICATIONS FOR EXTENSION OF COMPLETION TIME FOR AN INOPERABLE EMERGENCY DIESEL GENERATOR (TAC NOS. ME8893, ME8894)

Dear Mr. Hamrick:

The Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment Nos. 264 and 292 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for Brunswick Steam Electric Plant, Unit Nos. 1 and 2, respectively, in response to your letter dated June 19, 2012, as supplemented by letters dated January 21, May 14, and August 29, 2013, and January 22, 2014. The amendments extend the completion time for an inoperable emergency diesel generator from 7 to 14 days.

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

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

Sincerely,

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

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 264 to DPR-71
2. Amendment No. 292 to DPR-62
3. Safety Evaluation

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