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

June 7, 2017

Mr. William R. Gideon, Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, LLC
8470 River Rd. SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING REVISIONS OF TECHNICAL SPECIFICATIONS TO ADOPT TSTF-535, "REVISE SHUTDOWN MARGIN DEFINITION TO ADDRESS ADVANCED FUEL DESIGNS" (CAC NOS. MF8864 AND MF8865)

Dear Mr. Gideon:

The Commission has issued the enclosed Amendment No. 277 to Renewed Facility Operating License No. DPR-71 and Amendment No. 305 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The amendments are in response to your application dated November 18, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16343A521).

The amendments adopt the NRC-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs" (ADAMS Accession No. ML112200436), Revision 0, dated August 8, 2011, revising the Technical Specification definition of shutdown margin (SDM) to require calculation of SDM at the reactor moderator temperature corresponding to the most reactive state throughout the operating cycle (68 degrees Fahrenheit (°F) or higher). The purpose is to address the Brunswick Steam Electric Plant, Units 1 and 2, boiling-water reactor fuel designs, which may be more reactive at shutdown temperatures above 68 °F.

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

Sincerely,

Michael J. Wentzel, 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. 277 to
   License No. DPR-71
2. Amendment No. 305 to
   License No. DPR-62
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING REVISIONS OF TECHNICAL SPECIFICATIONS TO ADOPT TSTF-535, "REVISE SHUTDOWN MARGIN DEFINITION TO ADDRESS ADVANCED FUEL DESIGNS" (CAC NOS. MF8864 AND MF8865) DATED JUNE 7, 2017

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ADAMS Accession No.: ML17088A396 * By Memo

OFFICE	LPL2-2/PM	LPL2-2/LA	SRXB/BC *	STSB/BC(A)
NAME	MWentzel	BClayton	EOesterle	JWhitman
DATE	05/24/17	05/23/17	03/21/17	05/30/17
OFFICE	OGC (NLO)	LPL2-2/BC	LPL2-2/PM	
NAME	BHarris	UShoop (AHon for)	MWentzel	
DATE	06/05/17	06/07/17	06/07/17	

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 277
Renewed License No. DPR-71

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated November 18, 2016, 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 paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine S. Shoop, 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: June 7, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 277

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace Page 6 of Renewed Operating License DPR-71 with the attached Page 6.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.1-5	1.1-5

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

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; and is subject to all applicable provisions hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, are hereby incorporated in the license. Duke Energy Progress, LLC 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

1.1 Definitions (continued)

OPERABLE-OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2923 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

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UNITED STATES
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DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 305
Renewed License No. DPR-62

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated November 18, 2016, 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 paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine S. Shoop
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: June 7, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 305

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace Page 6 of Renewed Operating License DPR-62 with the attached Page 6.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.1-5	1.1-5

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

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) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 305, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Renewed Facility Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233. For SRs that existed prior to Amendment 233, 1.2 Definitions (continued)

OPERABLE-OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2923 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 277 AND 305

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

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated November 18, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16343A521), Duke Energy Progress, LLC (Duke Energy), the licensee, proposed changes to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant, (Brunswick or BSEP) Units 1 and 2. Specifically, the licensee requested to adopt U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STSs) Change Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs" (ADAMS Accession No. ML112200436), dated August 8, 2011.

The proposed change would revise the TS definition of shutdown margin (SDM) to require calculation of SDM at the reactor moderator temperature corresponding to the most reactive state throughout the operating cycle (68 degrees Fahrenheit (°F) or higher). The purpose is to address newer boiling-water reactor (BWR) fuel designs, which may be more reactive at shutdown temperatures above 68 °F. This license amendment request is consistent with NRC-approved TSTF Traveler TSTF-535.

2.0 REGULATORY EVALUATION

2.1 Background

In water-moderated reactors, water is used to slow down, or moderate, high energy fast neutrons to low energy thermal neutrons through multiple scattering interactions. The low energy thermal neutrons are much more likely to cause fission when absorbed by the fuel. However, not all of the thermal neutrons are absorbed by the fuel; a portion of them are instead absorbed by the water moderator. The amount of moderator and fuel that is present in the core heavily influences the fractions of thermal neutrons that are absorbed in each.

Water-moderated reactors are designed such that they tend to operate in what is known as an under-moderated condition. In this condition, the ratio of the moderator-to-fuel in the core is small enough that the overall effectiveness of water as a moderator decreases with increasing temperature; fewer neutrons are absorbed in the moderator due to the decrease in its density, but this is overshadowed by the reduction in the number of neutrons that moderate from high fission energy to the lower energy level needed to cause fission. The result is a decrease in power and temperature: a negative reactivity feedback effect where the reactor becomes self-regulating. However, if the amount of moderator becomes too large with respect to the amount of fuel, the reactor can enter an over-moderated condition. In this condition, the overall effectiveness of water as a moderator increases with increasing temperature; the reduction in the number of neutrons absorbed in the moderator outweighs the loss in neutrons reaching lower energies. This causes an increase in power that leads to a further increase in temperature creating a potentially dangerous positive reactivity feedback cycle.

As practical examples in support of the proposed changes to the definition of SDM, TSTF-535 discussed SDM with regards to GE14 and GNF2 fuels. TSTF-535 indicated that for historical fuel products through GE14, the maximum reactivity condition for SDM always occurred at a moderator temperature of 68 °F because these fuel products were designed so that the core is always under-moderated when all control rods are inserted, except for the single most reactive rod. In cores with GNF2 fuel, TSTF-535 stated that it is expected that the maximum reactivity condition at beginning of cycle will remain at 68 °F, but that later in cycle the most limiting SDM may occur at a temperature greater than this, indicating that with this fuel design the core could potentially achieve an over-moderated condition.

2.2 Technical Specification Changes

Duke Energy adoption of TSTF-535 for Brunswick Units 1 and 2 proposes to revise the TS definition of SDM to require calculation of SDM at the reactor moderator temperature corresponding to the most reactive state throughout the operating cycle (68 °F or higher).

The current definition of SDM in Section 1.1, "Definitions," of the Brunswick Units 1 and 2 TS is:

> SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:
>
> a. The reactor is xenon free;
>
> b. The moderator temperature is 68 °F; and
>
> c. All control rods are fully inserted except for the single control rod of the highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

The licensee proposes the following changes (shown in bold) to the definition of SDM in accordance with TSTF-535:

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical **throughout the operating cycle** assuming that:

a. The reactor is xenon free

b. The moderator temperature is ≥ 68 °F, **corresponding to the most reactive state**; and

c. All control rods are fully inserted except for the single control rod of the highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

2.3 Regulatory Review

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 26, "Reactivity control system redundancy and capability," and GDC 27, "Combined reactivity control systems capability," respectively require that reactivity within the core be controllable to ensure subcriticality is achievable and maintainable under cold conditions, with appropriate margin for stuck rods; and that reactivity within the core be controllable to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Among other things, 10 CFR 50.36(c)(2)(ii) requires the establishment of a limiting condition for operation (LCO) for 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. The TS definition of SDM and the LCOs placed on SDM serve, in part, to satisfy GDC 26 and 27 by ensuring there is always sufficient negative reactivity worth available to offset the positive reactivity worth of changes in moderator and fuel temperature, the decay of fission product poisons, the failure of a control rod to insert, and reactivity insertion accidents. Given this margin, the core can be held subcritical for conditions of normal operation, including anticipated operational occurrences.

Brunswick Unit 1 and Unit 2 were not licensed to the 10 CFR Part 50, Appendix A, GDC. The Brunswick equivalents of GDC 26 and 27 are described in the Brunswick Updated Final Safety Analysis Report (UFSAR), Sections 3.1.2.3.7 and 3.1.2.3.8, respectively.

The Brunswick UFSAR Section 3.1.2.3.7 addresses GDC 26 as follows:

> Each reactor unit contains two independent reactivity control systems of different design principles. Control of reactivity is provided by a combination of movable control rods, burnable poison, and reactor coolant recirculation system flow. These systems are able to accommodate fuel burnup, load changes, and long term reactivity changes. Reactor shutdown by the Control Rod Drive System is designed to be sufficiently rapid to prevent fuel damage limits from being exceeded during either normal operation or any operational transients. A Standby Liquid Control (SLC) System provides independent backup to the above reactivity controls.
>
> The reactivity control system is designed to provide sufficient reactivity compensation under conditions of normal operation to make the reactor always

> subcritical from its most reactive condition. Means are provided for continuously regulating the reactor core excess reactivity and reactivity distribution.

The Brunswick UFSAR Section 3.1.2.3.8 addresses GDC 27 as follows:

> The BSEP boiling water reactors (BWR) will never require the combined capability of a poison addition system in addition to a control rod system. The reactivity control during postulated accidents is entirely by means of the Control Rod System with appropriate margin for a stuck rod; the pattern of rod usage is normally controlled through the rod worth minimizer program of the on-line computer. The reactivity control is aided by the inherent negative power coefficient of reactivity. An SLC System is provided as an independent backup shutdown system to cover emergencies in the operational reactivity control system. This system is designed to maintain the reactor in a safe shutdown condition.

The staff concludes that a comparison of GDC 26 and 27 with the design features and controls used at Brunswick Unit 1 and Unit 2 shows that the difference does not alter the conclusion that the proposed change as presented in TSTF-535, Revision 0, and the associated model safety evaluation is valid for Brunswick Unit 1 and Unit 2, and therefore, acceptable.

The NRC's guidance for the format and content of licensee TSs can be found in NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4," April 2012 (ADAMS Accession No. ML12104A192).

Revision 3 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 4.3, "Nuclear Design," dated March 2007 (ADAMS Accession No. ML070740003), provides the procedures concerning the review of control systems and SDM to help ensure compliance with GDC 26 and 27.

3.0 TECHNICAL EVALUATION

3.1 Current Definition of Shutdown Margin

In BWR plants, the control rods are used to hold the reactor core subcritical under cold conditions. The control rod negative reactivity worth must be sufficient to ensure the core is subcritical by a margin known as the SDM. It is the additional amount of negative reactivity worth needed to maintain the core subcritical by offsetting the positive reactivity worth that can occur during the operating cycle due to changes in moderator and fuel temperature, the decay of fission product poisons, the failure of a control rod to insert, and reactivity insertion accidents. Specifically, Section 1.1, "Definitions," of the STSs defines SDM as the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that the reactor is (1) xenon free, (2) the moderator is 68 °F, and (3) all control rods are fully inserted except for the rod of highest worth, which is assumed to be fully withdrawn.

The three criteria provided in the definition help exemplify what has traditionally been the most reactive design condition for a reactor core. Xenon is a neutron poison produced by fission product decay and its presence in the core adds negative reactivity worth. Assuming the core is xenon free removes a positive reactivity offset and is representative of fresh fuel at the beginning of cycle. The minimum temperature the reactor moderator is anticipated to experience is 68 °F, making it the point at which the moderator will be at its densest and, therefore, capable of providing the highest positive reactivity worth. By assuming the highest worth rod is fully withdrawn, the core can be designed with adequate SDM to ensure it remains safely shutdown even in the event of a stuck control rod, as required by GDC 26 and 27.

Determination of the SDM under the aforementioned conditions yields a conservative result that, along with the requirements set forth in Section 3.1.1 of the TS, helps ensure:

a. the reactor can be made subcritical from all operating conditions and transients and design-basis events,

b. the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and

c. the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

3.2 Proposed Definition of Shutdown Margin

The specified moderator temperature of 68 °F facilitates the maximum reactivity condition only if the core exists in an under-moderated condition. In addition to burnable poisons, many modern fuel designs also incorporate partial length rods for increased neutron economy that are employed in order to extend the operating cycle. Both of these affect the ratio of moderator to fuel. The strong local absorption effects of the burnable poisons in fresh fuel make the core under-moderated. As burnable poisons are depleted during the fuel cycle, the core becomes less under-moderated, potentially leading to a slightly over-moderated condition wherein the core will be more reactive at a moderator temperature higher than the 68 °F specified in the SDM definition. Thus, the maximum core reactivity condition and the most limiting SDM may occur later in the fuel cycle at a temperature greater than 68 °F. Consequently, calculation of the SDM at the currently defined moderator temperature of 68 °F may not accurately determine the available margin.

TSTF-535, therefore, proposed a change to the definition of SDM to enable calculation of the SDM at a reactor moderator temperature of 68 °F or a higher temperature corresponding to the most reactive state throughout the operating cycle. SDM would be calculated using the appropriate limiting conditions for all fuel types at any time in core life.

In support of the proposed change, TSTF-535 cited the requirements for SDM as specified in Topical Report NEDO-24011-A, Revision 18, "General Electric Standard Application for Reactor Fuel (GESTAR II)," dated April 2011 (ADAMS Package Accession No. ML111120046). Section 3.2.4.1 of GESTAR II states:

> The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn and all other rods fully inserted.

The Traveler also cited SRP Section 4.3, which states the following concerning the review of control systems and SDM:

> The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation.

> The applicant shall discuss shutdown margins (SDM). Shutdown margins need to be demonstrated by the applicant throughout the fuel cycle.

Although the licensing basis requirements for SDM in GESTAR II are only applicable for cores licensed with Global Nuclear Fuel's methods, they are consistent with the review procedures set forth in the SRP, which are provided to help ensure compliance with GDC 26 and 27. TSTF-535 stated that while the SRP does not prescribe the temperature at which the minimum SDM should be determined, the requirement of shutting down the reactor and maintaining it in a shutdown condition "at any time during operation" suggests that considering a range of thermal and exposure conditions would be appropriate in the determination of the minimum SDM. Because newer fuel designs employ elements such as partial length rods and burnable absorbers, which may cause the maximum core reactivity conditions and the most limiting SDM to occur later in the fuel cycle at a temperature greater than 68 °F, the NRC staff agrees with the TSTF-535 assessment in this regard. Additionally, the NRC staff finds that allowing calculation of the SDM at the most limiting core reactivity condition is prudent with respect to ensuring compliance with GDC 26 and 27, or their plant-specific equivalent, and concludes that the proposed changes to the Brunswick TSs are acceptable.

The impetus for TSTF-535 was to provide for a more broadly applicable SDM definition in recognition of modern fuel designs, for which the core may not be in its most reactive condition at 68 °F. The proposed language would require the licensee to consider all temperatures equal to or exceeding 68 °F, and all times in the operating cycle. This change places an additional responsibility on any implementing licensee to identify the most limiting time-in-cycle and temperature, a change that is effectively more restrictive than the current definition. Therefore, the change can be considered acceptable for any facility that currently, and appropriately so, adheres to the current definition of SDM that is contained in the STSs. Based on these considerations, the NRC staff concludes that the change is applicable to any BWR, including Brunswick Units 1 and 2, regardless of its fuel design. The NRC staff also finds that the revised definition is consistent with the 10 CFR 50.36 requirements pertaining to LCOs, because it ensures that the LCOs for SDM consider a broadly conservative range of potential initial conditions in the anticipated operational occurrence analyses.

3.3 Summary

The NRC staff has reviewed the licensee's implementation of TSTF-535 proposed revisions to the definition of SDM. Based on the considerations discussed above, the NRC staff concludes that the proposed revisions are acceptable and will provide a conservative and improved approach to the calculation of SDM that ensures use of the appropriate limiting conditions for all current fuel types at any time in the life of the core. The NRC staff finds the proposed revisions serve to satisfy the requirements set forth in Brunswick-specific equivalents of GDC 26 and GDC 27, as discussed in NUREG-0800, Chapter 4.3, "Nuclear Design." Additionally, the NRC staff concludes the proposed changes to the definition of SDM would require the licensee to calculate SDM in consideration of the most limiting conditions in the core. Therefore, the revised SDM definition is acceptable for BWR facilities, including Brunswick, Units 1 and 2, using any current fuel design.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments on May 31, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (82 FR 4929). Accordingly, the amendments meet 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 amendments.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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