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

July 25, 2016

Mr. Benjamin C. Waldrep
Site Vice President
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road
M/C HNP01
New Hill, NC 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT ON MAIN STEAM SAFETY VALVE LIFT SETTING TOLERANCE, NOMINAL REACTOR TRIP SETPOINT ON PRESSURIZER WATER LEVEL, AND PRESSURIZER WATER LEVEL SPAN IN THE TECHNICAL SPECIFICATIONS (CAC NO. MF7195)

Dear Mr. Waldrep:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 151 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1, in response to Duke Energy Progress, Inc.'s (Duke Energy) application dated December 17, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15362A169), as supplemented by letters dated April 25, 2016, and June 8, 2016 (ADAMS Accession Nos. ML16116A285 and ML16161A359, respectively).

The amendment changes the technical specifications (TSs) to revise the (1) reactor trip setpoint on pressurizer water level-high percentage instrument span, (2) pressurizer water level span, and (3) as-found lift setting tolerance for the main steam line code safety valves, in TS Sections 2.2.1, "Limiting Safety System Settings Reactor Trip System Instrument Trip Setpoints"; 3.4.3, "Pressurizer"; and 3.7.1.1, "Turbine Cycle Safety Valves."

Duke Energy performed a new analysis for the overpressure evaluation of the Final Safety Analysis Report (UFSAR), Section 15.2.3, turbine trip event, as part of this license amendment request. The updated UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.71(e).

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.

If you have any questions concerning this matter, please contact me at 301-415-2760 or Martha.Barillas@nrc.gov.

Sincerely,

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 151 to NPF-63
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 151
Renewed License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, Inc. (the licensee), dated December 17, 2015, as supplemented by letters dated April 25, 2016, and June 8, 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 as 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. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 151, are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of issuance. The Updated Final Safety Analysis Report (UFSAR) changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Tracy J. Orf, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: July 25, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 151

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 4	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 marginal lines indicating the areas of change.

Remove	Insert
2-4	2-4
3/4 4-10	3/4 4-10
3/4 7-3	3/4 7-3

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

(1) Maximum Power Level

Duke Energy Progress, Inc. is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 151, are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, Inc. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

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[1] The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1.	Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2.	Power Range, Neutron Flux					
	a. High Setpoint	5.83	4.56	0	≤ 108% of RTP** See NOTES 7, 8	≤ 109.5% of RTP**
	b. Low Setpoint	7.83	4.56	0	≤ 25% of RTP** See NOTES 7, 8	≤ 26.8% of RTP**
3.	Power Range, Neutron Flux, High Positive Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5.	Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6.	Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7.	Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8.	Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10.	Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11.	Pressurizer Water Level-High	8.0	3.42	1.75	≤ 87% of instrument span See NOTES 7, 8	≤ 88.5% of instrument span

**RTP = RATED THERMAL POWER

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water level of less than or equal to 75% of indicated span, and at least two groups of pressurizer heaters each having a capacity of at least 125 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit power (kW) at least once per 18 months.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER			LIFT SETTING (± 3%)*	ORIFICE SIZE (IN.^{2})
STEAM GENERATOR				
A	B	C		
1MS-43	1MS-44	1MS-45	1170 psig	16.0
1MS-46	1MS-47	1MS-48	1185 psig	16.0
1MS-49	1MS-50	1MS-51	1200 psig	16.0
1MS-52	1MS-53	1MS-54	1215 psig	16.0
1MS-55	1MS-56	1MS-57	1230 psig	16.0

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, INC.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated December 17, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15362A169), as supplemented by letters dated April 25, 2016, and June 8, 2016 (ADAMS Accession Nos. ML16116A285 and ML16161A359, respectively), Duke Energy Progress, Inc. (Duke Energy, the licensee) submitted a request for changes to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Technical Specifications (TSs). The requested changes would revise the (1) reactor trip setpoint on pressurizer water level-high percentage instrument span, (2) pressurizer water level span, and (3) as-found lift setting tolerance for the main steam line code safety valves, in TS Sections 2.2.1, "Limiting Safety System Settings Reactor Trip System Instrument Trip Setpoints"; 3.4.3, "Pressurizer"; and 3.7.1.1, "Turbine Cycle Safety Valves." Additionally, Duke Energy performed a new analysis for the overpressure evaluation of the Updated Final Safety Analysis Report (UFSAR), Section 15.2.3, turbine trip event, as part of this license amendment request (LAR), and the amendment would update the UFSAR, Section 15.2.3, with the new analysis of record (AOR).

The supplements dated April 25, 2016, and June 8, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's initial proposed no significant hazards consideration determination as published in the *Federal Register* on April 5, 2016 (81 FR 19646).

2.0 REGULATORY EVALUATION

The main steam system supplies steam produced in the three steam generators (SGs) to the main turbine. The main steam system piping from the SGs up to the main steam isolation valves (MSIVs) is designed and fabricated to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Class 2. Each main steam line from the SG is provided with five spring loaded type main steam safety valves (MSSVs) that meet the requirements of ASME Code, Section III, Class 2, and Seismic Category I. The safety valves are designed to attain full lift at a pressure no greater than 3% above their set pressure, while maintaining the SG below the maximum allowable or less than 10% above the SG design pressure.

The HNP UFSAR, Section 5.2.2, "Overpressure Protection," discusses the requirements for overpressure protection. The reactor coolant system (RCS) overpressure protection during normal plant operation is accomplished by the utilization of pressurizer safety valves (PSVs), along with the reactor protection system and associated equipment. Combinations of these systems provide compliance with the overpressure requirements of the ASME Code, Section III, paragraphs NB-7300 and NC-7300 for pressurized-water reactor systems. Overpressure protection for the shell side of the SGs and the main steam line (MSL) up to the MSIVs is provided by the 15 SG safety valves (MSSVs), 5 on each MSL. The SG safety valve capacity is based on providing enough relief to remove 105% of the rated Nuclear Steam Supply System (NSSS) steam flow. This must be done by limiting main steam system pressure to less than 110% of the SG's shell side design pressure.

The reactor protection system provides an automatic reactor trip function to the reactor trip breakers to protect against unsafe and improper reactor operation during steady state and transient power operation and to provide initiating signals to mitigate the consequences of faulted conditions. The system uses input signals, including neutron flux, RCS temperature, RCS flow, pressurizer pressure, pressurizer level, SG level, reactor coolant pump under-voltage and under-frequency, turbine trip signals, and safety injection to provide a reactor trip signal. The pressurizer water level - high trip is designed to prevent rapid thermal expansion of the reactor coolant from filling the pressurizer. This trip function ensures a reactor trip is actuated prior to the pressurizer becoming water solid.

The proposed license amendment revises the as-found lift setting tolerance for MSL code safety valves (MSSVs) in TS 3.7.1.1, Table 3.7-2, from ± 1% to ± 3%. To support this MSSV setpoint tolerance change, changes are required to TS 2.2.1, Table 2.2-1. The reactor trip system instrumentation trip setpoint pressurizer water level-high percentage of the instrument is reduced from 92% to 87%. The allowable value of the instrument span is requested to be reduced from 93.5% to 88.5%. A change to reduce the maximum pressurizer water level limiting condition for operation (LCO) from less than or equal to 92% of indicated span to less than or equal to 75% of indicated, which requires a change to TS 3.4.3, is also proposed.

In support of these changes, the licensee seeks approval of new UFSAR, Section 15.2.3, turbine trip analysis, which has been performed to assess the impact of the requested changes on the transient and accident analyses included in the HNP UFSAR. Although the AOR is performed by AREVA Inc. (AREVA), the licensee has reanalyzed the turbine trip event for HNP using the RETRAN-3D computer code. The RETRAN-3D model for this analysis is based on U.S. Nuclear Regulatory Commission (NRC, or the Commission) approved methodology for the licensee's other plants (the Catawba Nuclear Station (Catawba), McGuire Nuclear Station (McGuire), and Oconee Nuclear Station (Oconee)). In this LAR, the licensee requests to apply the RETRAN-3D plant model to the HNP turbine trip event only. For this purpose, the licensee has performed and included the RETRAN-3D turbine trip benchmark against the existing AREVA turbine trip AOR. In its application, the licensee included an evaluation of the effect of the requested changes on the plant's UFSAR Chapter 15 events, and the results of its evaluation of other anticipated operational occurrences (AOOs) that are impacted by the proposed changes.

The NRC staff reviewed the licensee's application, as supplemented, against the following regulatory requirements and regulatory guidance documents.

2.1 Regulatory Requirements

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," the Commission established its regulatory requirements related to the TSs. The regulation in 10 CFR 50.36(a)(1) states, in part, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section."

Specifically, 10 CFR 50.36(c)(1)(ii)(A) states, in part:

> Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor.

Criterion 3 of 10 CFR 50.36(c)(2)(ii) requires that a TS LCO of a nuclear reactor be established for a structure, system, or component that is part of the primary success path and that functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Additionally, 10 CFR 50.36(c)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs [limiting conditions for operation] will be met."

The regulation in 10 CFR 50.46 (b)(1), "Peak cladding temperature," states that, "The calculated maximum fuel element cladding temperature shall not exceed 2, 200 °F [degrees Fahrenheit]."

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, establishes the minimum necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

General Design Criterion (GDC) 13, "Instrumentation and control," states that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions, as appropriate, to assure adequate safety, including those variables and systems that can affect the fission process, integrity of the reactor core, reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC 15, "Reactor coolant system design," states that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

GDC 20, "Protection system functions," states the protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC 31, "Fracture prevention of reactor coolant pressure boundary," states the RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

ASME Code, Section III, Article NB-7000, "Overpressure Protection," and Article NB-7311, Relieving Capacity of Pressure Relief Devices," specifies that the overpressure protection system provide sufficient relief capacity to prevent a pressure increase greater than 10% above the RCPB design pressure, accounting for loses through piping and other components.

2.2 Regulatory Guidance

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition," Section 5.2.2, Revision 3, "Overpressure Protection," provides guidance to determine whether the systems that provide overpressure protection to the Reactor Coolant Pressure Boundary satisfy the requirements of GDC 15, "Reactor coolant system design," and GDC 31, "Fracture prevention of reactor coolant pressure boundary," and will perform their intended functions during all plant operating and accident conditions.

Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," dated December 1999 (ADAMS Accession No. ML993560062) describes a method that the NRC staff finds acceptable for use in complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and will remain within, the TS limits. RG 1.105 endorses Part I of Instrument Society of America (ISA)-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," subject to NRC staff clarifications.

RG 1.105 states:

> In association with the trip setpoint and limiting conditions for operation (LCOs), the LSSS [limiting safety system setting] establishes the threshold for protective system action to prevent acceptable limits being exceeded during design-basis accidents. The LSSS therefore ensures that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 (ADAMS

Accession No. ML051810077), addresses requirements on limiting safety system settings that are assessed during the periodic testing and calibration of instrumentation.

By letter dated September 7, 2005, from Patrick L. Hiland (NRC) to the Nuclear Energy Institute Setpoint Methods Task Force, "Technical Specification for Addressing Issues Related to Setpoint Allowable Values" (ADAMS Accession No. ML052500004), the NRC specifies footnotes that should be added to surveillance requirements related to setpoint verification for instrument functions on which a safety limit has been placed. This letter also addresses the information that should be included within TSs to ensure operability of the instruments following surveillance tests related to instrument setpoints.

2.3 Supplemental Guidance

The Pressurized-Water Reactor and Boiling-Water Reactor Owner's Groups' Technical Specification Task Force Traveler (TSTF)-493, Revision 4, dated January 5, 2010 (ADAMS Accession No. ML100060064), and an errata sheet dated April 23, 2010 (ADAMS Accession No. ML101160026), address the NRC staff's concerns stated in RIS 2006-17.

On May 11, 2010, the NRC published a notice in the *Federal Register*, "Notice of Availability of the Models for Plant-Specific Adoption of Technical Specifications Task Force Traveler TSTF-493, Revision 4, 'Clarify Application of Setpoint Methodology for LSSS Functions'," (75 FR 26294), documenting its position on the adoption of TSTF-493, Revision 4.

3.0 TECHNICAL EVALUATION

3.1 Instrumentation Setpoint Change

The NRC staff reviewed the following TS changes related to Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints, Functional Unit 11, Pressurizer Water Level-High instrumentation":

1. Reduce Trip Setpoint from 92% to 87% of the instrument span.
2. Reduce the Allowable Value from 93.5% to 88.5% of the instrument span.
3. Add Footnotes 7 and 8 to Trip Setpoint which read as follows:

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument

> Setpoints." The as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report."

Within the LAR submittal, the licensee provided a summary of the setpoint calculation results in Attachment A, HNP-I/INST-1010, "Evaluation of RTS/ESFAS [Reactor Trip Setpoint/Engineered Safety Features Actuation System] Tech Spec Related Setpoints, Allowable Values, and Uncertainties," Table 3-8, for the pressurizer water level-high. The licensee made available for audit the full calculation HNP-I/INST-1010 for detailed setpoint calculations. The staff noted the methodology employed by the licensee was based on a Westinghouse setpoint methodology, which the staff had previously found to be acceptable for use on Westinghouse 2-loop, 3-loop, and 4-loop pressurized-water reactor plants.

In response to an NRC staff question regarding the assumptions and the sources of the inputs used for the HNP-specific application of this methodology, the licensee stated that the methodology in Reference 2.8 of the calculation, documented in Westinghouse Commercial Atomic Power (WCAP)-15249, Revision 0, "Westinghouse Protection System Setpoint Methodology for Harris Nuclear Plant" (a proprietary technical report), was used as the methodology basis for this calculation. The licensee also stated that Reference 2.9d from HNP-I/INST-1010, Westinghouse Calculation Note CN-SSO-99-5, Rev. 1, dated September 7, 1999, contained the input uncertainty data values used to support the determination of channel uncertainties for the pressurizer water level – high reactor trip setpoint uncertainty calculation.

The NRC staff noted that a complete copy of Revision 0 of Calculation HNP-I/INST-1010 had been submitted for evaluation as a response to a staff request for information in conjunction with its review of HNP's SG replacement and power uprate license amendment. This WCAP was transmitted to the staff in the licensee's letter dated May 18, 2001, from Mr. James Scarola, Vice President, HNP, to the U.S. NRC Document Control Desk (ADAMS Accession No. ML011450219). The staff also noted that the corresponding pages within the revision of the calculation applicable to the revised pressurizer high level trip setpoint for the current license amendment application are still shown as Revision 0.

Section 3.2, "Inputs and Assumptions," of calculation HNP-I/INST-1010 identifies in input 3.2.5 that, "Uncertainty components are defined using a 95% probability and high confidence level ..." and in input 3.2.13, "Rack temperature effects [RTE] are based upon historical Westinghouse performance data, and can be considered to reflect uncertainty values at 95% probability and 95% confidence level."

The staff notes that RG 1.105, which endorses, with exceptions and clarifications, the ISA Standard S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation," states that Section 4 of ISA-S67.04-1994 specifies the methods, but not the criterion, for combining uncertainties in determining a trip setpoint and its allowable values. The 95/95 tolerance limit is an acceptable criterion for uncertainties.

The licensee calculated the channel statistical allowance (CSA) (generally referred to in the industry as total loop uncertainty), as a sum-of-the-square-root of the independent uncertainty tolerances and the algebraic addition of the dependent uncertainty tolerances.

The staff evaluated the licensee's use of the uncertainty terms identified in the calculation and found that the licensee has used all of the terms appropriate to this Westinghouse setpoint methodology for this type of instrument channel, and that where a particular term does not apply, Westinghouse stated the uncertainty value was 0. The CSA calculated was 5.25% of span.

The licensee selected trip setpoint (TS) (generally referred to in the industry as nominal trip setpoint) as:

TS = ≤ 87.0% level span

The licensee stated the safety analysis limit (SAL) (generally referred to in the industry as analytical limit (AL)) as 95% of level span.

The licensee calculated the total allowance (TA) as equal to SAL- TS = 8% of span and the additional margin available as equal to TA – CSA = 2.75% span.

The NRC staff notes this available margin is approximately half the magnitude of the CSA value, providing additional assurance to offset the limitation in sensor/transmitter uncertainties not meeting the 95/95 acceptance criterion for uncertainties recommended in RG 1.105. This magnitude of margin helps ensure that the automatic protective action will correct the abnormal situation before a safety limit is exceeded, as stated in RG 1.105.

From data provided in the calculation, the licensee calculated allowable value (AV) as equal to 88.5% of level span.

The licensee selected the sensor as-left tolerance (ALT) as equal to the sensor reference accuracy (SRA) of ± 0.5% span.

The licensee calculated the sensor as-found tolerance (AFT) as follows, where SD stands for sensor drift and SMTE stands for sensor measurement and test equipment:

$$AFT = [(ALT)^2 + (SD)^2 + (SMTE)^2]^{1/2} = 1.46\%\ \text{span}$$

The licensee selected the rack as-left tolerance as equal to the rack reference accuracy of ± 0.5% span.

The licensee calculated rack as-found tolerance (AFT) as follows, where RD stands for rack drift:

$$AFT = [(ALT)^2 + (RD)^2 + (RMTE)^2]^{1/2} = 1.14\%\ \text{span}$$

The NRC staff approved the addition of Notes 7 and 8 for other instrument channels for this plant in conformance with TSTF-493, Option A, under HNP License Amendment No. 139 (ADAMS Accession No. ML11356A096). Notes 7 and 8 pertain to the use of the AFT and ALT for use in monitoring the performance of the instrument channel during successive surveillances. These notes were found to be consistent with the guidance in TSTF-493, Option A, for Westinghouse plants. At this time, the licensee is adding Notes 7 and 8 for the pressurizer water level-high instrument channel. The staff finds this TS change acceptable, since it is consistent with

TSTF-493. The staff notes the licensee has developed adequate plant procedures for ensuring compliance with the TSTF-493 notes.

Based on the information provided, the staff finds the licensee has performed the necessary setpoint calculations for the ALTs and AFTs, sensor, and rack uncertainties, total loop uncertainties, nominal trip setpoint, and allowable value in conformance with RG 1.105, TSTF-493, and RIS 2006-17. The staff has concluded that the methodology demonstrates that the proposed setting limits provide reasonable assurance that the automatic protective action will correct the abnormal situation before a safety limit is exceeded.

The staff also finds the licensee has plant surveillance procedures enabling compliance with TSTF-493. Specifically, the licensee stated its surveillance procedure requires the setpoint to be returned to within the specified as-left calibration tolerance if found outside the ALT band. If the setpoint is found outside the AFT band, plant surveillance procedures require an evaluation before returning the channel to service. Hence, the proposed TS changes comply with the requirements of 10 CFR 50.36; GDC 13; GDC 20; and 10 CFR 50, Appendix A. Therefore, the proposed changes to Functional Unit 11, pressurizer water level-high, in TS Table 2.2-1 are acceptable.

Based on its review of the calculations and materials submitted with the licensee's application, the NRC staff concludes that although the licensing commitment for the plant is to RG 1.105, Revision 1, the licensee's setpoint calculations are consistent with RG 1.105, Revision 3. This is acceptable to the staff. The addition of surveillance notes into the TS, in accordance with the guidance of TSTF 493, will ensure instrument channels will be functioning as required and that actions taken to ensure operability will be administered properly.

The NRC staff also concludes that the revised protective channel setting will continue to meet the criteria in 10 CFR 50, Appendix A, GDC 13, because the licensee's methodology for selecting the nominal trip setpoint enables the safety-related instrument channels to perform safety actions, while remaining functionally capable of monitoring variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. This is because the methodology establishes setpoints that are conservative with respect to the safety analysis limits and accounts for the instrument channel performance uncertainties present under such conditions. The revised setting also meets the criteria in 10 CFR 50, Appendix A, GDC 20, because the licensee's methodology for selecting the nominal trip setpoint ensures that setpoints for safety-related instrument channels will initiate the operation of appropriate systems at a value conservative to the safety analysis limit.

Further, the licensee's methodology satisfies the requirements of 10 CFR, Part 50; 10 CFR 50.36(c)(1)(ii)(A); and 10 CFR 50.36 (c)(3) because the setting derived using this methodology accounts for all anticipated uncertainties such that the protective action will correct the abnormal situation before a safety limit is exceeded. Also, the methodology provides for the establishment of AFT values that are used to identify whether automatic protective instrument channels are functioning as required. During periodic surveillances required by TSs of the performance of these instrument channels, if it is determined that the protection channel setting operation during steady state and transient power operation and to provide initiating signals to igate the consequences of faulted conure safety limits are not exceeded. Also, the channel surveillance process allows the licensee to meet the requirements of 10 CFR 50.36(c)(3), because such surveillances adopting the AFT values determined by the analysis methodology assure that the necessary quality of the instrument channel is maintained, facility operation will remain within safety limits, and the LCOs will be met.

3.2 Overpressure Analysis

The function of the MSSVs is to maintain the SG pressure within limits appropriate for safe operation, including assuring that the maximum pressure does not exceed safety limits during all AOOs. The MSSVs are located outside of the containment between the containment penetration and the MSIV. The MSSV capacity is based on providing enough relief to remove 105% of the rated steam flow. This must be done by limiting main steam system pressure to less than 110% of the SG shell side design pressure. To avoid lifting during pressure transients, set pressures for the MSSVs are as high as possible within the code requirements. To prevent chattering during operation of the MSSVs, the individual valves in each steam supply line are set at different lift pressures.

The licensee requests approval to change the tolerance from ±1% to ±3%, as shown in Table 1. However, the implementation of the change results in higher tolerance for the actuation pressure of the MSSVs.

Table 1: HNP, Unit 1, MSSVs Requested Set Pressure Change (psig)

Valve Number*			**Nominal Set Pressure**	**Set Pressure at +1% Tolerance**	**Set Pressure at +3% Tolerance**
Steam Generator					
A	**B**	**C**			
1 MS-43	1 MS-44	1 MS-45	1,170	1,181.7	1,205.1
1 MS-46	1 MS-47	1 MS-48	1,185	1,196.9	1,220.6
1 MS-49	1 MS-50	1 MS-51	1,200	1,212.0	1,236.0
1 MS-52	1 MS-53	1 MS-54	1,215	1,227.2	1,251.5
1 MS-55	1 MS-56	1 MS-57	1,230	1,242.3	1,266.9

* Design pressure of 1,200 per square inch absolute (psia), design temperature of 600 °F, accumulation of 3%, and orifice size of 16 in^2 apply to all valves.

The higher MSSVs setpoints cause less heat transfer from the primary system and higher primary pressure. This results in less high pressure safety injection flow into the system. To evaluate the effect of higher MSSV setpoints on the RCS and SG, first the AOOs that would result in the higher pressurization of the primary system and the secondary side of the SGs are identified. Second, the events that result in the highest peak cladding temperature (PCT) are determined.

The NRC staff verified that (a) all of the affected AOOs associated with the proposed LAR changes have been identified; (b) NRC-approved methodologies have been used, along with conservative inputs and assumptions to determine the limiting AOOs; and c) the results of the limiting AOOs remain below the safety limits, with reasonable margin to satisfy the requirement of the applicable GDC. In particular, RCS pressure, SG pressure, PCT, the departure from nucleate boiling ratio (DNBR), and piping stresses remain below their related safety limits.

Regarding the limiting AOOs with respect to RCS and SG pressure, events that result in the primary-side and secondary-side pressurization include turbine trip, loss of main feedwater, rod cluster control assembly withdrawal, loss of coolant flow, loss of non-emergency alternating current (AC) power, and seized reactor coolant pump rotor. Among these events, a turbine trip (an American Nuclear Society Condition II event, fault of moderate frequency) without a concurrent reactor trip is the most limiting AOO with respect to the RCS and SG pressurization. A turbine trip may be the result of a loss of electric load, loss of condenser vacuum, a spurious reactor trip, low turbine oil pressure, etc. Therefore, the licensee has reanalyzed this event using the higher pressure setpoint for MSSV actuation to ensure that the calculated RCS and SG pressures remain below the corresponding 110% design pressure limits. In this event, reactor trip is initiated by the reactor protection system due to high pressurizer pressure and not upon turbine trip. The conservative set of inputs and assumptions for the RCS pressurization analysis differs from the set of conservative inputs and assumptions utilized in the SG pressurization analysis. For example, the pressurizer pressure control system (pilot-operated relief valve [PORV] and spray) is not credited in the RCS pressurization analysis (so that RCS pressure rapidly reaches the SRV setpoint). The pressure control system is credited in the SG pressurization analysis (to conservatively prolong time to reactor trip). A third case is also analyzed to evaluate the effect on the fuel design limit, which may be approached due to core thermal-hydraulic conditions leading to departure from nucleate boiling (DNB). In this case, credit is taken for the function of any system that leads to RCS depressurization (to conservatively reduce margin to DNB).

With respect to PCT, a small-break loss-of-coolant accident (SBLOCA) is the most limiting event. In the SBLOCA transients, secondary pressure rises to the MSSV setpoint upon reactor/turbine trip and remains there until a primary phase change at the break occurs, which is associated with higher energy release from the primary system. The effect of a higher set pressure for MSSV actuation on PCT is due to a higher SG average temperature, as the saturation temperature is a direct function of pressure. The drop in temperature gradient across the SG tubes results in a lower rate of heat transfer from the RCS to the secondary side of the SG. This, in turn, leads to a higher RCS pressure and lower safety injection flow rate. The maximum fuel rod cladding temperature will also increase, assuming no change in the decay heat rate.

The NRC staff confirmed that inputs were consistent with the limitations and conditions (L&Cs) in the RETRAN-3D code the licensee used for the turbine trip analysis. The licensee benchmarked the RETRAN-3D turbine trip model against the AREVA AOR for the turbine trip analysis and obtained agreement for the transient sequence of events, the system parameter responses, and the peak primary and secondary pressure results. The staff's independent analysis confirmed the applicant's results. The TRACE confirmatory analysis, as discussed in this safety evaluation report, focuses only on system overpressurization.

The effect of the increase in the MSSV as-found lift setting tolerance, and the auxiliary feedwater (AFW) flow rate changes due to increases in secondary side pressure, on the SBLOCA is explained in the UFSAR, Section 15.6.5.3, and supporting plant calculations. This UFSAR section and supporting calculations were reviewed for inputs, assumptions, and the analysis results with respect to absolute value and trend. The analysis using a conservative set of inputs and assumptions calculates a 32 °F increase in the PCT. However, despite the +32 °F PCT impact, the 10 CFR 50.46(b) PCT criterion of 2,200 °F continues to be met with sufficient margin.

The turbine trip analysis is performed by the licensee using the RETRAN-3D code to evaluate the effect of the MSSV as-found lift setting tolerance increase on system pressurization. The licensee has also performed a DNB analysis, which uses a statistical core design method to avoid performing a cycle-specific analysis. The statistical method determines the most limiting set of inputs and boundary conditions to calculate the limiting departure from nucleate boiling ratio (DNBR). The sensitivity analysis shows that the DNBR is insensitive to the proposed change in MSSV setpoint.

The disposition of the Chapter 15 non-loss-of-coolant accident (LOCA) events for the planned MSSV tolerance changes from ±1% to ±4% (the LAR seeks only ±3%) relative to the current MSSV tolerance is divided into two categories. One category includes events such as loss of feedwater, loss of external electrical load, reactor coolant pump shaft seizure, and feedwater pipe break. The disposition of these events is performed by AREVA. AREVA's disposition was reviewed with special attention to assumptions and the results of the conclusions. As a result of the NRC staff's review, it was concluded that a correct approach had been used in the disposition process. The summary of the tabulated results of AREVA's disposition is also in Table 2 of the licensee's LAR dated December 17, 2015. The second category includes events such as turbine trip, SG tube rupture, anticipated transient without scram, and the radiological events. These events are analyzed by the licensee, which has determined that a turbine trip is the most limiting event regarding the planned MSSV tolerance changes.

The licensee used RETRAN-3D to analyze the turbine trip event. The AOR for the licensee's UFSAR Chapter 15 AOOs is provided by AREVA calculations. The licensee's LAR also seeks approval of using the RETRAN-3D code for turbine trip analysis for HNP. The methodology used in the analysis is based on the licensee's following topical reports:

1. DPE-NE-3000-PA, Revision 5a, Oconee, McGuire, and Catawba Nuclear Stations, "Thermal-Hydraulic Transient Analysis Methodology," dated October 2012 (Enclosure 4 to ADAMS Accession No. ML16032A004); and

2. DPE-NE-3002-A, Revision 4b, "UFSAR Chapter 15 System Transient Methodology," dated September 2000 (ADAMS Accession No. ML003756662).

These references have been previously reviewed and approved by the NRC for the licensee's Catawba and McGuire stations and describe methodologies to analyze system transients for using RETRAN-3D. The NRC safety evaluation report on RETRAN-3D dated January 25, 2001 (ADAMS Accession No. ML010470342), imposed 45 L&Cs. To seek approval for the use of RETRAN-3D for the turbine trip analysis for HNP, the licensee has reviewed each of these imposed L&Cs and has provided explanations for the resolution of each L&C. For this evaluation, the licensee divided the L&Cs into two categories. The first category consists of those L&Cs that are either dispositioned in the same manner as the previously-approved methodologies for Catawba and McGuire or are not applicable to the HNP turbine trip analysis and are, therefore, dropped from further consideration. The second category consists of a total of eight L&Cs, which the licensee dispositioned because their disposition had changed from the previously-approved methodologies. The NRC staff reviewed the licensee's evaluation and dispositions and finds them acceptable. The staff verified that the methodology was used in the same manner as the approved methodology used for Catawba and McGuire, was consistent with the approval of RETRAN-3D, or was only changed in a minor way that would not impact the system overpressurization, and is, therefore,

acceptable for the proposed new AOR for the turbine trip. Accordingly, the remaining L&Cs are satisfied. As a result, the RETRAN-3D analysis is approved by the NRC staff for use by the licensee to update the HNP UFSAR, Chapter 15.2.3, turbine trip analysis, based on the (a) prior approval of a similar methodology for Duke's other PWRs, (b) licensee's satisfactory benchmark of the RETRAN-3D results against the AOR, and (c) the staff's independent analysis, which confirmed the applicant's results.
aThe NRC staff performed a confirmatory analysis as part of the audit, using the TRACE computer
The NRC staff performed a confirmatory analysis as part of the audit, using the TRACE computer
code. The NRC staff requested input data that included SG design data, reseat pressures for the pressurizer SRVs, PORVs, secondary side MSSVs, and fuel type(s) with axial and radial power shapes for its confirmatory analysis.

In its supplement dated April 25, 2016 the staff requested that the applicant verify that the turbine trip is the most limiting event. In particular, the applicant reviewed events such as loss of external electrical load, inadvertent closure of the MSIVs, loss of condenser vacuum, and other events resulting in turbine trip, loss of non-emergency AC power to the station auxiliaries, loss of normal feedwater flow, and feedwater system pipe break. Those results indicated that the turbine trip event experiences the fastest and most complete loss of secondary side heat removal for the events listed in the HNP UFSAR, Section 15.2, "Decrease in Heat Removal by the Secondary System." In running the TRACE model, the staff observed that the turbine trip event causes a rapid pressure excursion on both primary and secondary systems such that system pressures temporarily overshoot the safety valve lift setpoints. The loss of electrical load, inadvertent closure of the MSIVs, loss of condenser vacuum, and other events resulting in a turbine trip, are similar to the turbine trip event but are less limiting, since the initiating events isolate the secondary side from the turbine heat sink at a slower rate. For these events, the loss of secondary side heat removal is less dynamic, leading to a slower primary and secondary system pressurization.

The licensee stated that the requested changes to the MSSV lift setpoint tolerance will not invalidate the following conclusions in the UFSAR. The primary and secondary system overpressure results for the revised turbine trip analysis are similar to, or bound, the values from the previous UFSAR, Section 15.2.3, turbine trip overpressure AOR. As a result, the revised UFSAR, Section 15.2.3, turbine trip overpressure results continue to bound the overpressurization results for other ANS Conditions II, III, and IV events, including, but not limited to, the UFSAR events listed above. Since all limiting event determinations previously described in the HNP UFSAR remain valid, the NRC staff concludes that the licensee's reanalysis demonstrated that the revised turbine trip analysis continues to bound the individual events listed in Table 2 of the LAR.

In its supplement dated April 25, 2016, the licensee explained the reason for the change in the PSV setpoint uncertainty in the turbine trip analysis and its impact on the calculation results. The licensee stated that the nominal opening setpoint for the PSVs is 2,485 pounds per square inch gauge (psig), with an uncertainty of ±1% per HNP TS 3.4.2. The licensee performed a new analysis for the overpressure evaluation of the UFSAR, Section 15.2.3, turbine trip event, modeling the PSV opening setpoint, assuming an uncertainty of +3%. With this assumption, the PSVs open at 2,560 psig (i.e., 2,485 psig +3%) compared to 2,510 psig (i.e., 2,485 psig +1%). The uncertainty assumed when modeling the PSV opening setpoint, regardless of whether it is +1% or +3%, has no effect on the peak secondary pressure calculation because the peak pressurizer pressure remains below 2,485 psig. Increasing the opening setpoint of the PSVs delays the opening of the PSVs, which yields a more limiting result for the peak primary pressure evaluation. While the applicant used +3% in the analysis, the applicant did not request an increase in the PSV setpoint tolerance, which remains at 1%. Confirmatory calculations have shown that the peak pressure on the secondary side is not significantly affected by the lift setpoint of the PSV for either the primary overpressurization case or the secondary overpressurization case. The pressurizer pressure does not exceed the setpoint (2,485 psig) during the secondary overpressurization case, so there is no change to the peak secondary side pressure.

To ensure the turbine trip is the most limiting event, in its supplement dated April 25, 2016, the licensee stated that the dispositions of the HNP UFSAR accident analyses, summarized in Table 2 of the licensee's LAR, are obtained by dividing the HNP UFSAR accident analyses into five categories and then providing the reason why events in each category are less limiting than a turbine trip. These categories include the following:

1. UFSAR events that do not involve an NSSS transient. There is no impact on these events, as the components and systems affected by the requested changes are not affected by the requested change.
2. UFSAR events where an evaluation dispositions the event against a more limiting UFSAR event.
3. UFSAR events where the components and systems affected by the requested changes are not exercised in the AOR.
4. UFSAR events where the primary and/or secondary system pressure response in the AOR is affected by the requested changes, but the overpressure results are bounded by the revised UFSAR, Section 15.2.3, turbine trip analysis.
5. UFSAR events where additional discussion or evaluation is required to investigate the impact of the requested changes.

The NRC staff finds it acceptable to use this comprehensive approach because the licensee evaluated the impact of the proposed changes on all of the UFSAR events and properly dispositioned these events to determine the most limiting AAO.

The TRACE code was used to perform confirmatory calculations of the turbine trip event. Two sets of conditions were utilized for two types of analyses: primary system overpressurization and secondary system overpressurization. The assumptions as to which safety systems are available differ slightly for the primary overpressurization case and for the secondary overpressurization case. The following assumptions apply for both cases. The same initial conditions for each case were used in the confirmatory analysis as were used in the licensee's analysis. The methodology for modeling the opening of SRVs (MSSV and PSV) in the confirmatory calculations was also adapted from the licensee's analysis. The pressurizer relief valves are assumed to be operational. It is conservatively assumed that the reactor is in manual control; otherwise, the control rod banks would insert prior to a trip, and the severity of the transient would be reduced. No credit is taken for the steam dump system or SG PORVs. The main feedwater flow into the SGs is assumed to be lost at the time of the turbine trip, and auxiliary feedwater flow is also assumed to be inoperable. It is assumed conservatively that the reactor does not trip on a turbine trip, because a reactor trip at the time of turbine trip would reduce the severity of the transient.

The primary overpressurization case does not take credit for the pressurizer spray and pressurizer PORVs, since those devices would reduce the severity of the transient.

The secondary overpressurization case assumes that the pressurizer spray and pressurizer PORVs are operational during the transient. This limits the reactor coolant pressure and increases the time that the reactor operates before a scram signal is reached, therefore, increasing the severity of the transient.

The primary system overpressurization initial conditions were used for two transients – one that allows for a pressurizer trip on high pressurizer pressure, and the other that assumes that the high pressurizer pressure trip is inactive. The two primary system overpressurization and one secondary overpressurization transients were evaluated, assuming +3% uncertainty on the MSSV setpoint. The results of these three calculations are summarized below.

Table 2 shows an event summary for the turbine trip primary overpressurization case with the high pressurizer pressure trip active. A comparison is made between the results of the +3% confirmatory calculations and the HNP RETRAN-3D calculations of the LAR dated December 17, 2015.

Table 2: Event Summary for Turbine Trip with Pressurizer High Pressure Trip

Events	Time (sec)	
	+3% HNP (Ref. 1)	+3% TRACE
Turbine trips	0.00	0.00
Pressurizer high pressure trip signal reached	5.07	5.00
PSVs open	6.90	8.64
Reactor trips on pressurizer high pressure (rod motion starts)	7.07	7.08
Peak primary pressure at bottom of reactor vessel reached	7.84	8.74
PSVs close	9.40	10.19
Bank 1 MSSVs open	12.90	10.08
Bank 2 MSSVs open	14.00	11.42
Bank 2 MSSVs close	NA	38.60†
Bank 1 MSSVs close	NA	45.91†
End of simulation	60.00	60.00

† Bank 1 and Bank 2 MSSVs close in the TRACE analysis.

It is noted that the first and second banks of MSSVs close in the confirmatory calculation but not in the HNP calculations in the LAR dated December 17, 2015. It is also noted that the events directly connected to the primary system occur later in the transient (notably the PSV actions and vessel peak pressure). Additionally, the MSSVs open somewhat earlier in the confirmatory calculation when compared to the HNP results. Most notably, the MSSVs close before the end of the transient in the TRACE simulation. This, coupled with the lack of a second bank of MSSVs opening, shows that there is less energy being transferred from the primary to the secondary system in the confirmatory calculation. The HNP RETRAN-3D calculation is, therefore, more conservative than the TRACE confirmatory calculation.

The primary system overpressurization without high pressurizer pressure trip case tests the results when the trip on high pressure in the pressurizer is not credited in the reactor protection logic. Otherwise, the initial conditions and assumptions are the same as those for the primary system overpressurization case discussed previously. It is expected that the reactor scram will now occur due to a trip on high pressurizer level. Table 3 summarizes the timing of key events in this analysis.

Table 3: Primary Overpressurization Without Pressurizer High Pressure Trip Case

Event	Time (s)	
	3% HNP (Ref. 1)	+3% TRACE
Turbine trips	0.00	0.00
High pressurizer level trip signal reached	NA	5.42
Reactor trips on high pressurizer level (control rod motion starts)	NA	7.48
PSVs open	6.90	8.64
Peak primary pressure at bottom of reactor vessel reached	7.85	8.86
Bank 1 MSSVs open	NA	10.08
PSVs close	9.50	10.35
PSVs open	10.00	†
High pressurizer level trip signal reached	10.91	NA
Bank 1 MSSVs open	12.70	NA
Reactor trips on high pressurizer level (control rod motion starts)	12.91	NA
Bank 2 MSSVs open	13.60	11.36
Bank 3 MSSVs open	14.90	‡
PSVs close	16.30	†
Bank 4 MSSVs open	17.10	‡
Bank 4 MSSVs close	39.00	‡
Bank 3 MSSVs close	41.70	‡
Bank 2 MSSVs close	47.30	36.84
Bank 1 MSSVs close	NA	43.84
Auxiliary feedwater (AFW) on lo-lo level	58.30	*
End of simulation	60.00	60.00

† PSVs do not open a second time in the TRACE analysis.
‡ Bank 3 and Bank 4 MSSVs open in the HNP analysis but remain closed in the TRACE analysis.
* No auxiliary feedwater is actuated in the TRACE analysis.

The MSSVs in the current TRACE confirmatory calculation do not respond in the same way as the MSSVs do in the HNP RETRAN calculations. The pressure in the secondary side is not as high as in the HNP calculations, which results in only two of the MSSV banks opening during the transient. For the secondary system overpressurization case, the pressurizer PORVs and the spray are conservatively active, where for the primary-side overpressurization case, they were not. This helps to keep the pressure in the primary side down, which is conservative for the case of secondary-side pressurization. Table 4 summarizes the timing of the events during this turbine trip transient.

Table 4: Secondary Side Overpressurization Case

Event	Time (s)	
	+3% HNP (Ref. 1)	+3% TRACE
Turbine trips	0	0.00
Pressurizer spray initiates	0	NA
Pressurizer compensated and non-compensated PORVs open and cycle	2.4	2.22
Bank 1 MSSVs open	5.4	3.23
Bank 2 MSSVs open	6.2	3.93
High pressurizer level trip signal reached	NA	4.64
Bank 3 MSSVs open	7.3	5.22
Reactor trips on high pressurizer level (rod motion starts)	NA	6.78
Pressurizer non-compensated and compensated PORVs close	NA	8.31
High pressurizer level trip signal reached	9.34	NA
Pressurizer spray initiates	NA	10.01
Bank 4 MSSVs open*	10.1	NA
Reactor trips on high pressurizer level (rod motion starts)	11.34	NA
Bank 5 MSSVs open*	14.6	NA
Pressurizer non-compensated and compensated PORVs close	15.2	NA
Peak secondary pressure occurs at bottom of the SG down comer	17.3	11.23
Bank 5 MSSVs close	33.4	NA
Bank 4 MSSVs close	35.3	NA
Bank 3 MSSVs close	38.3	24.80
Bank 2 MSSVs close	46.7	27.29
Bank 1 MSSVs close	NA	35.78
AFW on lo-lo SG level	57.9	-
End of simulation	60	60.00

* Bank 4 and Bank 5 MSSVs open in HNP's RETRAN-3D but remain closed in the TRACE analysis.

The comparison of peak pressures for both primary and secondary-side pressurizations, obtained by the licensee's RETRAN-3D code and the NRC staff's TRACE code for +3% tolerance, is shown in Table 5.

Table 5: Comparison of Calculated Peak Pressures (psia) – For + 3% Case

System Region	Pressure (psia)		
	HNP (LAR dated 12/17/15)	TRACE	Limit
Primary side – bottom of vessel	2,740	2,738	2,750
Secondary side – bottom of SG downcomer	1,305	1,277	1,320

Comparing the results obtained from the TRACE confirmatory analysis with those obtained by the licensee's RETRAN-3D, demonstrates the following:

1. The NRC staff's evaluation of both the licensee's RETRAN and TRACE analyses show similar trends in the overall system response to a turbine trip.

2. Regarding the absolute values, the peak pressures for both primary and secondary side pressurizations calculated by TRACE, as a best estimate code, are lower than those calculated by RETRAN. Therefore, the results calculated by the licensee are conservative and provide adequate support for the subject LAR.

The NRC staff determined that the proposed changes to TSs 2.2.1, 3.4.3, 3.7.1.1, and the UFSAR, Section 15.2.3, are acceptable for HNP. This determination is based on the TRACE confirmatory analyses demonstrating that the licensee's analyses to support the proposed changes in the as-found MSSV lift setpoints are conservative. The calculations with the TRACE code confirmed that the peak system pressure for the limiting turbine trip event will not exceed 110% of the design pressure, in accordance with ASME Code, Section III, Article NB-7000. The licensee's RETRAN-3D analysis for the HNP turbine trip event was successfully benchmarked by the licensee against the AOR results. In addition, the TRACE independent confirmatory analysis performed by the NRC yielded results consistent with RETRAN-3D. The RETRAN-3D methodology used by the licensee is based on the licensee's approved topical reports referenced above. These references have been previously reviewed and approved by the NRC for the licensee's Catawba and McGuire stations. The NRC's safety evaluation report dated January 25, 2001, imposed 45 L&Cs that the licensee reviewed and dispositioned. The NRC staff reviewed the licensee's evaluation of each L&C and found the licensee's dispositions acceptable. Hence, the staff confirmed that the 45 L&Cs imposed by the January 25, 2001, safety evaluation report are met. The licensee's RETRAN-3D analysis demonstrates that the TS LCO proposed in the LAR will protect the integrity of the reactor pressure vessel with the PSV LCO in place. Accordingly, the staff finds the requirements of 10 CFR 50.36(c)(2)(ii), Criterion 3, are met.

The NRC staff concludes that since the licensee has met the applicable regulatory requirements discussed above, the transient-specific changes to the UFSAR based on RETRAN-3D analyses and the TS changes proposed in the LAR satisfy the requirements of GDC 15 and GDC 31 that the overpressure protection system maintains RCS pressure within acceptable design limits and with sufficient margins during normal operation and AOOs.

4.0 STATE CONSULTATION

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

5.0 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 (81 FR 19646; April 5, 2016). 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.

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.

Principal Contributors: Fred Forsaty
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David Rahn

Date: July 25, 2016

If you have any questions concerning this matter, please contact me at 301-415-2760 or Martha.Barillas@nrc.gov.

Sincerely,

/RA/

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 151 to NPF-63
2. Safety Evaluation

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NAME	MBarillas	BClayton (LRonewicz for)	EOesterle	MWaters	DAlley (JTsao for)
DATE	06/23/16	07/07/16	06/30/16	06/03/16	06/24/16
OFFICE	DSS/STSB/BC	DSS/SNPB/BC*	OGC – NLO w/comments	DORL/LPL2-2/BC(A)	DORL/LPL2-2/PM
NAME	AKlein	JDean	BHarris	TOrf	MBarillas
DATE	06/24/16	06/30/16	07/05/16	07/22/16	7/25/16

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