May 24, 2016

MEMORANDUM TO: Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Operation

FROM: Eric R. Oesterle, Acting Chief ***/RA/***
Reactor Systems Branch
Division of Safety System
Office of Nuclear Reactor Regulation

SUBJECT: AUDIT REPORT - APPLICATION FOR AMENDMENT SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1. THE PROPOSED LICENSE AMENDMENT REVISES THE AS-FOUND LIFT SETTING TOLERANCE FOR MAIN STEAM LINE CODE SAFETY VALVES IN TECHNICAL SPECIFICATIONS (TAC NO. MF7195)

By letter dated December 17, 2015 (ADAMS Accession No. ML153A169), Duke Energy (the licensee) submitted a license amendment request (LAR) for the Shearon Harris Nuclear Power Plant, Unit 1. The proposed license amendment revises the as-found lift setting tolerance for Main Steam Line Code Safety Valves (MSSVs) in Technical Specification (TS) 3.7.1.1, Table 3.7-2, from ± 1% to ± 3%. To support the MSSV setpoint tolerance change, changes are required to TS 2.2.1, Table 2.2-1. The U. S. Nuclear Regulatory Commission staff conducted a regulatory audit to increase the efficiency of the LAR review. The audit report is attached.

Enclosure:
Audit Report

CONTACT: Fred M. Forsaty, NRR/ DSS
(301) 415-8523

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Audit Report

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Accession Number: ML16138A050 NRR-106

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AUDIT REPORT - APPLICATION FOR AMENDMENT SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1. THE PROPOSED LICENSE AMENDMENT REVISES THE AS-FOUND LIFT SETTING TOLERANCE FOR MAIN STEAM LINE CODE SAFETY VALVES IN TECHNICAL SPECIFICATIONS (TAC NO. MF7195)

BACKGROUND

By letter dated December 17, 2015 (ADAMS Accession No. ML153A169), Duke Energy (the licensee) submitted a license amendment request (LAR) for the Shearon Harris Nuclear Power Plant (HNP), Unit 1. The proposed license amendment revises the as-found lift setting tolerance for MSSVs in Technical Specifications, from ± 1% to ± 3%.

AUDIT REPORT

The U. S. Nuclear Regulatory Commission (NRC) staff conducted a regulatory audit of the licensee to increase the efficiency of the LAR review. The U. S. Nuclear Regulatory Commission (NRC) staff focused on the following topics during this audit:

Obtaining the following information for the NRC staff to perform confirmatory analysis using TRACE code:

1) Steam Generator (SG) design information including, but not limited to:

- General geometrical – elevation of major components, volumes, lengths, flow areas,diameters, etc.
- Elevation of the lowest/average/highest U-tubes
- Secondary side liquid volume vs. elevation
- Recirculation ratio
- Heat structures – material, volume, thicknesses, surface area, etc.
- Number of plugged tubes – minimum and maximum
- Location of level instrumentation (narrow and wide range)
- Nominal operating conditions – Dome pressure, secondary side mass/level, feedwater flow (FW) flow,
- FW temperature, primary side flow, primary side inlet/outlet temperatures, DP through tubes, etc.

2) Reseat pressures for the pressurizer safety regulator valves (SRVs), pressure operated relief valves (PORVs) and secondary side (SRVs) to model hysteresis effects).

Enclosure

3) Fuel type(s) along with axial and radial power shapes. (Note: Some information is available in the FSAR, however, it shows data for several fuel types; LOPAR, VANTAGE 5 and AREVA 17x17 designs. The FSAR does not include power shapes so we will need that information.)

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- For the analyses listed in Table 2 of the LAR, audited the applicable margins before and after the proposed change.
- Discussions related to the events listed on Table 2 of the LAR. Discussed whether there are any differences in the input conditions or user-specified fuel thermal properties for the events determined to be bounded by the analysis of record (AOR) or bounded by one of the reanalyzed events.
- Discussed the limiting analysis listed in Table 2, prior and after the proposed change listed in the LAR.
- Discussed to support the information provided in Sections 3.3.3 to 3.3.10 of the LAR.
- Discussed regarding the revised TS Bases B 2-3
- Discussed related to the following calculations:

  1) Feedwater Line Break
  2) Turbine Trip
  3) Limiting analysis applicable to this LAR

- Discussed possible restrictions related to S-RELAP5 analysis methods that could impact the requested amendment. Discussed the extent of the NRC approval being requested by this LAR. Discussed how the licensee has exercised its responsibility to ensure that AREVA adheres to the conditions in the SEs and the limitations stated in the Topical Reports for the codes and methodologies used in analyses in the proposed LAR.

To evaluate the HNP LAR, the licensee also provided all the required references and documents, on a secured website for the NRC and ISL (NRC Contractor) review as a part of this audit. The major documents that were audited are:

<u>AREVA DOCUMENT</u> – AREVA calculation package, “Harris SBLOCA Sensitivity Study for MSSV Tolerance increase”, (32-9236498-000) dated April 2015 was reviewed for inputs, assumptions, and the analysis results with respect to absolute value and trend. The analysis documented in this calculation package was to assess the increase of the MSSV tolerance on the Small Break Loss of Coolant Accident (SBLOCA) Peak Cladding Temperature (PCT).

<u>AREVA DOCUMENT</u> – AREVA calculation package, “Harris PSV and MSSV Tolerance Increase Disposition of Non-LOCA Events”, was reviewed with special attention to inputs, assumptions, and the results for the analysis with respect to absolute value and trend. This calculation package documents the disposition of the transient response to Chapter 15 events for the planned MSSV tolerance changes from ±1% to ±4% (the LAR seeks only ±3%) relative to the current MSSV tolerance. The disposition is limited to the system response to the particular event that affect the system transient. The licensee has analyzed other aspects such as Turbine Trip event, Steam Generator Tube Rupture and ATWS events, and the radiological events. The

AREVA evaluation concludes that no system reanalysis by AREVA is needed to support cycle operation.

AREVA DOCUMENT – This calculation package (SHA1-20 OPDT setpoint Analysis documents the Over Power Delta-T (OPDT) reactor trip setpoint verification analysis performed for the HNP operating cycle 20. The confirmation of the OPDT trip setpoints is a standard analysis performed every cycle. This trip is designed to protect against fuel centerline melt (FCM) during normal operation, operational transients, and AOOs. The OPDT trip setpoints ensures, with a 95% probability at a 95% confidence level, that no location in the core experiences FCM.

AREVA DOCUMENT – The analysis in this calculation package (SHA1-20 OTDT Setpoint Analysis) describes the calculations to verify the Over-Temperature Delta Temperature (OTDT) setpoint correlations for the HNP Cycle 20. The analysis estimates the one-sided, statistically adjusted minimum 95/95 margin between the OTDT trip power and the power to cause departure from nucleate boiling (DNB) and hot leg saturation. These power margins take into account the cycle specific fuel and system changes for Harris Nuclear Plant Cycle 20. Revision 1 remains valid.

AREVA DOCUMENT - This calculation package (Shearon Harris (SHA1) CSLL Analysis for MSSV Tolerance Increase, Safety Related, Revision 1.0) is performed to verify the Core Safety Limit Lines (CSLLs) for the HNP. The analysis in this calculation package supports the operation of Cycle 20 with increased MSSV lift setpoint tolerances. In addition to the increase in the MSSV setpoint tolerance, this analysis considers a mixed core penalty of two percent in order to be consistent with current plant conditions. The CSLLs are a combination of the Departure from Nucleate Boiling (DNB) limit lines and saturation temperature limit lines. This analysis confirms the Technical Specification CSLL limit.

SIEMENS/AREVA DOCUMENT - This calculation package (Harris Power Uprate Turbine Trip Analysis, E-6924-595-7) contains the analysis of a turbine trip event for conditions representing the power uprate and steam generator replacement {SGR/PUR). Three cases are analyzed in this document, each challenging a different safety criterion namely, the primary-side over-pressurization, the secondary over-pressurization, and minimum departure from nucleate boiling ratio (MDNBR). In addition the system pressurization cases, a sensitivity analysis is performed in which the pressurizer level is biased upward by the instrument uncertainty. The results will be captured in the HNP FSR upon NRC's approval of the licensee's LAR

DUKE DOCUMENT - This document includes the Technical Specification Basis and consists of two parts including: - DUKE ENERGY PROGRESS, INC., NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY, DOCKET NO. 50-400, SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, RENEWED FACILITY OPERATING LICENSE, Renewed License No. NPF-63. - Technical Specifications, Shearon Harris Nuclear Power Plant, Unit No. 1,Docket No. 50-400, Appendix "A" to License No. NPF-63 . All 404 pages of the bases document were visually scanned with special attention paid to Sections 2.1.2, 2.2.1, 3/4.4.3, and 3/4.4.4 regarding the changes that the licensee has proposed in the Enclosure 4 of the Harris LAR, "SERIAL HNP-15-038, ENCLOSURE 4, PROPOSED TECHNICAL SPECIFICATION BASES CHANGESSHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 DOCKET NO. 50-400.

DUKE DOCUMENT - This document includes the Shearon Harris Nuclear Power Plant FSAR Reviewed Section 15.2.3, Pages 15.2.3-1 through 15.2.3-9 for inputs and assumptions especially the itemized parameters in Table 15.2.3-1 (Page 15.2.3-5), Table 15.2.3-2 (Page 15.2.3-6), Table 15.2.3-3 (Page 15.2.3-7), and Table 15.2.3-4 (Page 15.2.3-8). The results reported in Table 15.2.3-5 (Page 15.2.3-8), Table 15.2.3-6 (Page 15.2.3-9) were reviewed for trend and absolute values. Although FSAR Section 15.2.3 is for ±1 tolerance of MSSVs, most of the input data and assumption are generic in nature. These set of input and assumptions were checked against the RETRAN-3D analysis for consistency.

DUKE DOCUMENT - This document includes the licensees analysis of HNP Turbine Trip event, “Harris FSAR Section 15.2.3 Turbine Trip and Safety Valve Setpoint Tolerance Analyses”, DPC-1552.08-00-0296, Dated September 2015. The inputs and assumptions of Tables 10 (page 42) Table 11 (page 44), and Table 13 (page 45) were reviewed for adequacy of degree of conservatism.

DUKE DOCUMENT - SERIAL HNP-15-038, ATTACHMENT C, “Harris Turbine Trip Methodology Qualification”, SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, DOCKET NO. 50-400, RENEWED LICENSE NUMBER NPF-63

The staff discussions during the audit resulted in a set of RAIs (ML16091A006) that were issued in March of 2016.