NRR-PMDAPEm Resource

From: Hon, Andrew
Sent: Wednesday, October 19, 2016 8:32 AM
To: Murray, William R. (Bill) (Bill.Murray@duke-energy.com) (Bill.Murray@duke-energy.com)
Subject: Audit Plan for Brunswick MELLLA+ License Amendment Request Review

REGULATORY AUDIT PLAN FOR BRUNSWICK STEAM ELECTRIC PLANT
UNIT NOS. 1 AND 2
TO SUPPORT REVIEW OF THE LICENSE AMENDMENT REQUEST
REGARDING CORE FLOW OPERATING RANGE
EXPANSION MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS
DOCKET NOS. 50-325 AND 50-324

BACKGROUND

By letter dated September 16, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16257A410), Duke Energy (the licensee) submitted a License Amendment Request (LAR) Regarding Core Flow Operating Range Expansion for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2 (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)).

The U.S. Nuclear Regulatory Commission (NRC) staff is performing a detailed review of the proposed changes. Due to the complexity of the proposed LAR, supporting calculations, and computer based modelling, the staff has determined that continuing interactions with licensee's technical staff can resolve complex technical issues more quickly than several rounds of Request for Additional Information questions requiring followup responses from the licensee.

The NRC staff has determined that a Regulatory Audit of the background technical analysis for this LAR should be conducted in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits," for the NRC staff to gain a better understanding of the licensee's calculations and other aspects of the LAR. This audit includes, but is not limited to, the gathering and transmittal of information to enable the NRC Staff to develop a TRACE model which will aid in the NRC staff's review. Additional informational needs for the NRC staff to gain a better understanding of other aspects of the LAR will be communicated as necessary throughout the duration of the audit.

REGULATORY AUDIT BASIS

A Regulatory Audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. A Regulatory Audit is conducted with the intent to gain understanding, to verify information and/or to identify information that will require docketing to support the basis for the licensing or regulatory decision.

INFORMATION NEEDS

Plant information described in this document in a searchable electronic format (e.g. PDF) provided on a compact disk. Note, in lieu of providing a response to each of the line items, modeling calculation notebooks and related documents which contain the information can be provided. These documents need not be provided all at once, they can be provided as they become available to licensee to transmit.

As necessary, make available technical staff or contractors who are familiar with the information to assist the NRC staff during the audit.

Additional information needs identified during the audit will be communicated to the designated point of contact.

TEAM ASSIGNMENTS/RESOURCE ESTIMATES

The duration of this audit is consistent with the LAR review schedule of October 2016 to September 2018. The resource estimate for this audit is approximately 300 hours of direct audit effort. The NRC staff performing this audit will include, but may not be limited to, the following staff:

AUDIT TEAM

- Diego Saenz , Audit Team Lead, Reactor Systems Branch Technical Reviewer, NRR
- Andy Hon, Brunswick Project Manager, Plant Licensing Branch II-2, NRR
- Eric Oesterle, Reactor System Branch Chief, NRR
- Kathy Gibson, Reactor Safety Engineer, Office of Nuclear Regulatory Research
- Shie-Jeng Peng, Reactor Systems Engineer, NRR
- Josh Borromeo, Reactor Systems Engineer, NRR
- Other reviewers, as needed.

LOGISTICS

The audit will start after the entrance phone call on October 19, 2016. Based on the need, the NRC staff will obtain clarification of the information by means such as phone calls, video conferencing, or site visits.

DELIVERABLES

At the conclusion of the audit, the NRC staff will conduct an exit briefing and provide a summary of audit results. The NRC staff plans to prepare a Regulatory Audit summary within 90 days of the completion of the audit.

PLANT INFORMATION NECESSARY FOR AUDIT

The following information will support the NRC staff conducting TRACE modeling in support of its review of the planned Brunswick LAR for an extension of the operating domain (i.e., MELLLA+):

Modeling Data Set 1

Please provide bundle nuclear design information for the staff to perform confirmatory cross-section calculations including:

a. A description of all lattices in the reference core design for MELLLA+, including pin arrangements and dimensions, channel dimensions, and gap dimensions
b. A description of the axial arrangement of all lattices in the bundle design(s)
c. The mass of uranium in each fuel type and in the overall bundle design(s)
d. Identification of the plant lattice (e.g. C- or D-, etc.)
e. Complete material composition description of all fuel assembly materials (structural material, spacers, cladding, fuel) including percentages of all isotopes
f. A description of the reference control blade design appropriate for performing lattice calculations, including materials and geometry
g. Tables of k-inf and fission rate density distributions as a function of exposure for various density histories

Modeling Data Set 2

Please provide the reference MELLLA+ core nuclear design information for the staff to perform confirmatory kinetics calculations including:

a. A description of the radial fuel loading by bundle design type
b. Tables that provide the nodal exposure and void history information for various points of interest during the cycle (e.g. Peak Hot Excess (PHE) and end of full power at minimum flow)
c. Provide critical control blade patterns for each point of interest in cycle. When describing the blade patterns please explicitly clarify any index relative to the top or bottom of active fuel and the length(s) of the active section(s) of the control blade
d. Core average direct energy deposition factors to inter- and intra-assembly bypass flows, active flows, and cladding
e. Tables of nodal powers for each exposure point

Modeling Data Set 3

Please provide the fuel thermal-mechanical design and analysis information for the staff to perform confirmatory thermo-mechanical calculations including:

a) A description of each unique fuel rod type in each fuel bundle design, including:

    a) Cladding dimensions, alloy, heat treatment, and surface roughness
    b) Pellet dimensions including any dish or chamfer information
    c) Heights of axial segments with differing pellet enrichments and additives (if applicable)
    d) Plenum dimensions
    e) Pellet characteristics (e.g. fraction of theoretical density, open porosity, surface roughness, expected density increase due to resintering, sintering temperature, etc.)
    f) The initial gas gap pressure and gas composition (if other than helium)
    g) A description of the plenum spring

b) A description of the power history assumptions used to derive parameters for subsequent transient analyses (e.g. the assumed linear heat generation rate history assumed for the average fuel rod and the peak power rod to determine gap conductance in downstream transient evaluations, and the axial power shapes assumed over the course of irradiation).

Modeling Data Set 4

Please provide the fuel bundle thermal-hydraulic design information for the staff to perform confirmatory thermal-hydraulic calculations including:

a. A description of the inlet and outlet nozzles of the fuel bundle, in particular the flow areas, heights, and local losses
b. The positions of fuel grid spacers for each bundle design

c. The axially dependent active flow areas and hydraulic diameters
d. The dimensions of any internal water channel(s)
e. The locations of the internal water channel(s) inlet and outlet
f. Water rod inlet and outlet loss coefficients
g. A description of the channel box geometry
h. Cladding surface roughness
   Counter current flow limiting locations and correlation parameters applicable to the fuel bundle design(s)
i. A description of hydraulic tests and measurements of pressure losses and critical bundle power
j. Results of critical power and local and bundle pressure drop measurements

<u>Modeling Data Set 5</u>

Please provide the reference MELLLA+ core thermal-hydraulic design information for the staff to perform confirmatory thermal-hydraulic calculations including:

a. A description of leakage path ways between the bundle and bypass, including loss coefficients and flow areas
b. The normal bypass flow fraction
c. The distribution of inlet orifices and orifice loss coefficients

<u>Modeling Data Set 6</u>

Please provide information regarding the plant parameters such as a plant parameters document or related documentation that gives detailed information of the Reactor Coolant System and associated systems appropriate for use in a Thermal-Hydraulic Systems analysis at MELLLA+ condition. This information should include (but is not limited to):

a. A plant heat balance diagram indicating major flows, pressures, and enthalpies
b. A description of initial or nominal conditions in terms of pressure, feed flow, recirculation flow, feed temperature, and level. If there are allowable ranges during normal operation (e.g. feed temperature ranges) provide these ranges as well.
c. A description of the reactor pressure vessel that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
d. A description of the control rod guide tubes that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
e. A description of the core support plate that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
f. A description of the upper tie plate that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
g. A description of the separators that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, losses, and carryover/carryunder fractions
h. A description of the dryer that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
i. A description of the upper head that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
j. A description of the downcomer that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
k. A description of the recirculation lines that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
l. A description of the recirculation pumps that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, losses, inertia, rated torque, rated speed, and homologous curves

m. A description of valves in the recirculation line (e.g. recirculation discharge and suction isolation valves and the recirculation flow control valve) that includes the geometry, locations of component interfaces, flow areas, hydraulic diameters, stroke time, and losses
n. A description of the jet pumps that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
o. A description of the main feedwater lines that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
p. A description of the steam lines that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, and losses
q. A description of the safety relief valves that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, stroke time, losses, and capacities
r. A description of the turbine bypass valves that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, stroke time, losses, and capacities
s. A description of the turbine control valves that provides the geometry, locations of component interfaces, flow areas, hydraulic diameters, stroke time, and losses
t. A description of the main steam isolation valve that provides geometry, locations of component interfaces, flow areas, hydraulic diameters, stroke time, and losses

u. A description of setpoints for various safety system functions, this should include:

    a. Narrow and Wide Range Level setpoints, e.g. Level 1, Level 2, Level 8, etc.
    b. Safety Relief Valve (SRV) lift and set pressures, tolerances, and opening/closing times
    c. High pressure trip setpoints
    d. Any automatic trip setpoints for the standby liquid control system

v. A description of the thermal properties of the vessel and associated internals, for example, the heat capacity and mass of the structures
w. A description of the containment wetwell and drywell, including;

    a. Initial pressure, temperature, and humidity (including allowable ranges during normal operating conditions)
    b. Initial wetwell liquid inventory, temperature, and pressure (including allowable ranges during normal operating conditions)
    c. Vacuum breaker characteristics
    d. Residual heat removal cooling capacity
    e. Volumes
    f. Passive heat structures

x. A description of the injection systems credited during Anticipated Transient Without Scram (ATWS) (e.g. control rod drive cooling, reactor core isolation cooling, and standby liquid control system) in terms of any automatic initiation signals, delays, flow rates, capacities, net positive suction head limitations, natural boron equivalent concentration, and sources (e.g. Condensate Storage Tank (CST), or wetwell)
y. A description of the three element level controller including the sensing locations for steam and feed flows as well as any and all gains and lags
z. A description of the electro-hydraulic control system in terms of turbine bypass and control valve operation

aa. The heat capacity temperature limit curve

Modeling Data Set 7

Please provide information regarding manual operator actions during postulated ATWS events. The description should provide timing information (e.g. operators begin reducing water level at X seconds), “trip point” information (e.g. once the temperature in wetwell reaches Y, the operators depressurize the vessel), and strategy information (e.g. to maintain water level at TAF+X). These actions may include:

a. Manual action to increase control rod drive cooling flow
b. Manual action to reduce reactor water level
c. Manual action to recover reactor water level
d. Manual action to initiate the standby liquid control system injection
e. Manual action to actuate the automatic depressurization system

Modeling Data Set 8

Please provide a sample systems analysis code input deck for the ATWS calculations and associated documentation (e.g. a User's Manual or Calculation Notebook) to assist the staff in ascertaining input values from the input deck.

Modeling Data Set 9

Please provide the following information pertaining to the reactor pressure vessel:

a. Total free volume of the reactor vessel.
b. Total water mass in the vessel during normal operation.
c. Fluid volume in each of the following key vessel internal components: lower head, bottom of core support region, core region, upper shroud dome region, standpipe region, separator region, dryer region, and downcomer region. Please also provide top and bottom axial elevations for each provided volume to clarify the region being described.

d. Top fuel guide grid dimensions and wall thickness.
e. Wall thickness of the upper shroud dome.
f. Curvature radius of the curved portion of the upper shroud dome, or alternatively the height and free volume inside this portion.
g. The total number, width, height, and thickness of shroud support legs in the lower plenum.
h. A vessel engineering drawing similar to Figures 5-3 and 5-4 in the Updated Final Safety Analysis Report (UFSAR), but of a size and clarity such that the dimensions are readable.
i. The magnitude of bypass flow from the lower plenum to the core bypass.

Modeling Data Set 10

Please provide the following information pertaining to the Reactor Coolant Pump:

a. The pump's effective moment of inertia.
b. The elevation of the Reactor Coolant Pump (RCP) with respect to the inside bottom of the reactor vessel.
c. A description of the recirculation loop which includes pipe lengths and orientations; flow areas and pipe wall thicknesses or pipe sizes; and any available information on friction losses.

Modeling Data Set 11

Please provide the diameter and wall thickness of the inlet riser pipe of the reactor jet pumps.

Modeling Data Set 12

Please provide the following information pertaining to the core channels:

a. Channel bypass hole size and number of holes.
b. Initial core loading map (fuel assembly type distribution) for Units 1 and 2, separately.
c. Assembly power distribution for Units 1 and 2, separately.
d. A fuel burnup (exposure) distribution for Unit 1, similar to that provided for Unit 2 in Table 4-1 of the UFSAR.

e. The radial power peaking factor in a fuel rod.

Modeling Data Set 13

Please provide the following information pertaining to the steam separators in the reactor vessel:

a. Number of stages in the separators.
b. Geometric data for each stage (wall radius, “pickoff” (liquid return) ring radius (see Figure 1), flow area, discharge diameter, barrel length, vane position, discharge loss, and effective L/D, or information from which these could be calculated).
c. Hub inlet radius.
d. Swirl vane angle relative to the horizontal (degrees).

e. Free volumes for each separator (or in total) inside the standpipe, the conical (swirl vane) section, the inner cylinder of each stage, and the liquid return section for each stage.
f. Arrangement (e.g., triangular or square lattice) and separator pitch.

Modeling Data Set 14

Please provide the following information pertaining to the steam dryers in the reactor vessel:

a. Inner diameter wall thickness, and bottom elevation of the dryer seal skirt.
b. Free volume inside the steam dryer region.
c. Average heat transfer area for all steam dryers in total.
d. Total metal mass and number of units in the steam dryer assembly.
e. Flow area and hydraulic diameter in the radial flow direction, or alternatively the total number of holes and hole diameter.

Modeling Data Set 15

Please provide the following information pertaining to the Control Rod Guide Tubes:

a. Flow area through the Control Rod Guide Tubes (CRGTs) near the lower core support plate, accounting for the presence of assembly nosepieces and control blades.
b. Free volume within a guide tube.

Modeling Data Set 16

Please provide the following information pertaining to the reactor for modeling of its point-kinetics behavior:

a. Average neutron lifetime.
b. Fuel temperature reactivity coefficient (fuel temperature vs. reactivity coefficient).
c. Coolant temperature reactivity coefficient (coolant temperature vs. reactivity coefficient).
d. Gas volume fraction reactivity coefficient (gas volume fraction vs. reactivity coefficient).

Modeling Data Set 17

Please provide the following information pertaining to the primary containment and suppression pool:

a. At what elevation relative to the top of the drywell foundation mat could spill-over occur between the drywell and the pressure suppression vent pipes?
b. UFSAR Table 6-3 provides the free volume of the drywell as 164,100 ft^3. Does this volume include the air space inside the pressure suppression vent pipes and/or downcomers? If not, please provide the normal (or minimum/maximum) air volume inside these elements, or sufficient geometric data or diagrams from which it could be calculated.
c. At what elevation above the bottom of the drywell is the SRV discharge to the suppression pool?

Modeling Data Set 18

Please provide the following information pertaining to the emergency core cooling systems and containment cooling:

a. UFSAR Section 12.2 lists the capacity of each CST as 500,000 gallons. Please clarify the actual nominal usable water volume in each CST, if it is different from this value.
b. What is the nominal temperature of water in the condensate storage tanks?
c. At what elevation inside the reactor vessel is the sparger for injection of flow from the Low-Pressure Core Spray system?
d. Please provide the steam flow rate and steam exit pressure for the turbine of the High-Pressure Coolant Injection pump.
e. Please provide the steam flow rate and steam exit pressure for the turbine of the Reactor Core Isolation Coolant (RCIC) pump.
f. Please provide a pump characteristic curve (developed head versus volumetric flow rate) for the RCIC pump.
g. As described in Sections 5.4.7, 6.2.2, and 6.3.2.2.4, the Residual Heat Removal/Low- Pressure Coolant Injection (RHR/LPCI) system is used for both low-pressure coolant injection and containment cooling. It may be envisioned that a demand for both functions could exist simultaneously, due to high suppression pool temperature while an LPCI signal has been generated. Please clarify which function takes precedence in such a case, or, if both functions are served at the same time, how the flow is apportioned between the two.
h. As described in Section 6.2.2, the containment cooling function of the RHR system can discharge flow to the suppression pool (via the full flow test line) or to spray headers located in the wetwell or drywell. Please clarify how the set point of discharge is decided, whether by automatic action or procedure. If containment cooling can split flow between two or more of these destinations simultaneously, please describe how the total flow is apportioned between them.

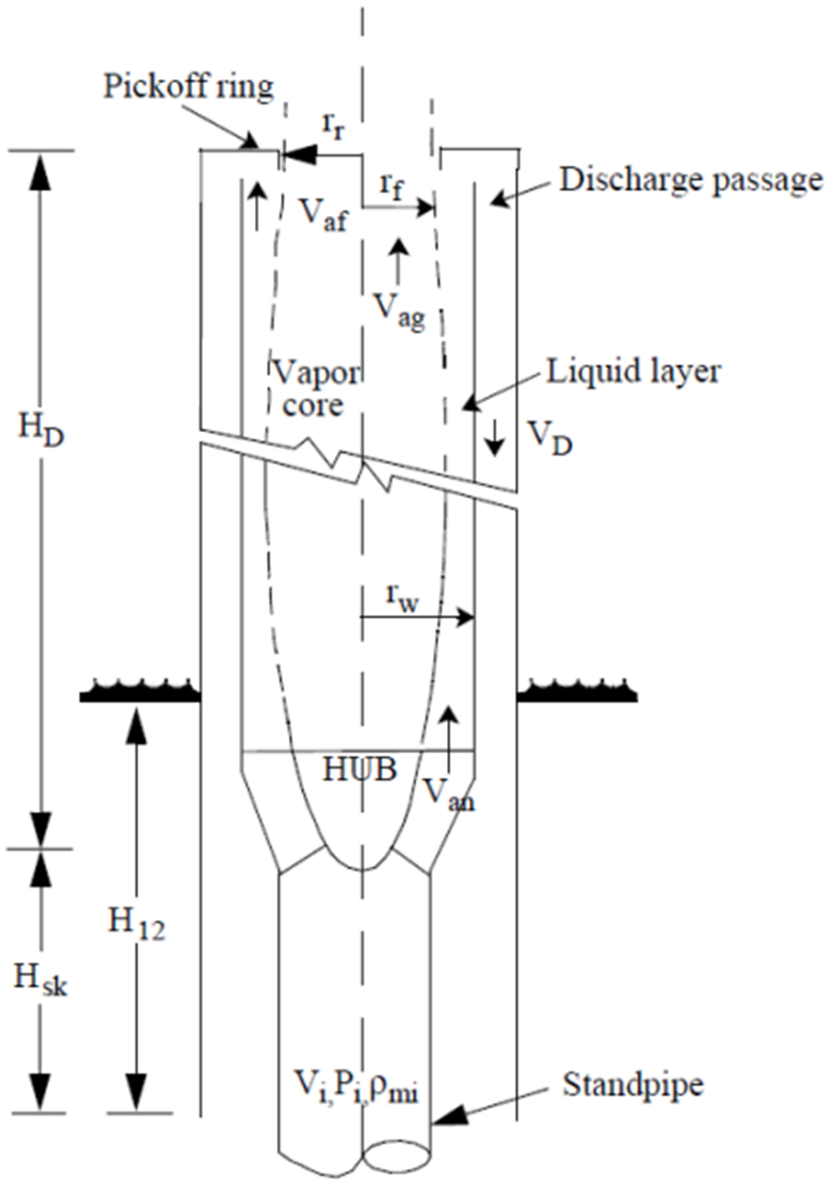

Figure 1 Schematic of first stage of mechanistic separator (Figure 10-13 [1])

[1] "TRACE V5.840 Theory Manual," Division of Safety Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, March 2013 (date of last revision).

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Mail Envelope Properties (Andrew.Hon@nrc.gov20161019083100)

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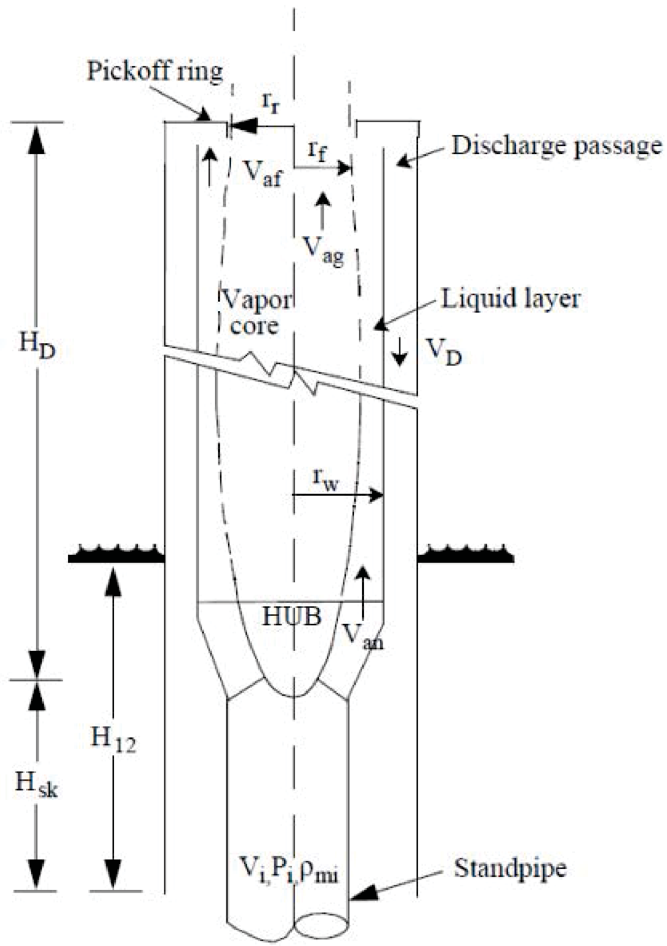
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