Southern Nuclear

Farley Pre-Submittal Meeting to Revise Technical Specification 5.5.17 to Increase Calculated Peak Containment Pressure

October 2021

Meeting Purpose and Agenda

- The purpose for this meeting is to discuss proposed amendment request to revise Technical Specification 5.5.17, “Containment Leakage Rate Testing Program” to Increase Calculated Peak Containment Pressure

- This meeting will cover the following topics:
  - Background
  - Proposed License Amendment Request
  - Analysis
  - Plant Impacts of Change
  - Precedent

Background

Background

- The containment, in conjunction with engineered safety features, is designed to withstand the internal pressure and coincident temperature resulting from the energy release of the LOCA associated with 2831 MWt and to limit the site boundary radiation dose to within the guidelines set forth in 10 CFR 100.
  - the design pressure limit of the containment is 54 psig
  - the design temperature limit is 280 °F

- A Containment Leakage Rate Testing Program has been established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions.
  - peak calculated containment internal pressure for the design basis loss of coolant accident, P_a
  - maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day

Proposed License Amendment Request

Proposed License Amendment Request

Reason for the change

- The change reflects updated calculations of the Loss of Coolant Accident (LOCA) mass and energy releases and corresponding containment responses. These calculations address issues raised in various Westinghouse Nuclear Safety Advisory Letters (NSALs). These NSALs identified modeling and material property errors in the analyses for the mass and energy releases. Consequently, the errors affected the predicted containment responses.
  - NSAL 14-2 "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties,"
  - NSAL 11-5 "LOCA Mass and Energy Release Calculation Issues"
  - NSAL 06-6 "LOCA Mass and Energy Release Analysis"

Technical Specification 5.5.17

Current TS 5.5.17

- The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

Proposed TS 5.5.17

- The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 45 psig .

Analysis

Analyses Needed

- Mass and energy (M&E) release calculation
  - To address the change described by the NSALs
- Containment pressure analysis
  - Uses input from the mass and energy release calculation to determine containment internal pressure

Mass and Energy Release Calculation Method

- WCAP-10325-P-A describes the NRC approved method for computing M&E releases into containment for a LOCA.
  - Models the short-term blowdown, refill, reflood, and long-term reflood mass and energy releases
  - Decay heat and stored energy are considered in the method

- SNC's revised LOCA M&E release calculation uses the WCAP method consistent with that used in the current licensing basis. The inputs are revised to reflect the modeling and material property errors identified in the NSALs. These analyses are performed in the same manner as the current analysis of record described in the Farley Updated Final Safety Analysis Report.

Mass and Energy Release Calculation

- The following changes to the method of evaluation were updated for the M&E release calculation revision:
  - Treatment of inactive metal mass
  - Turning on the drift flux model in the SATAN-IV code during the blowdown phase
  - Turning on the break flow with inertia model in the SATAN-VI code during the blowdown phase
- The drift flux model and the break flow with inertia model have been previously reviewed and approved for use within the conditions and limitations specified by the NRC.

- The refueling water storage tank (RWST) maximum water temperature used as an input to the M&E release analysis was lowered from 120 °F to 110 °F.
  - Makes RWST temperature input agree with the value used in the containment response analysis, and in agreement with plant design inputs

Containment Pressure Calculation Method

- The containment pressure analyses were performed using the GOTHIC computer program (version 8.1) .
- Two reactor coolant system breaks were analyzed. They are the limiting breaks for determining containment pressure.
  - Double Ended Hot Leg (DEHL) break - results in the higher peak pressure that occurs in the short-term
  - Double Ended Pump Suction (DEPS) break - the minimum safety injection case results in the overall limiting condition (which occurs in the long-term)

Containment Pressure Calculation

- The following design inputs were updated to reflect the current state of the plant:
  - The post-LOCA service water pond temperature calculation is revised. A bounding value of 107°F, which is present only at the end of the 30 days is conservatively used for the entire LOCA transient as input into the residual heat removal/ component cooling water model in the containment evaluation model.
  - The containment fan cooler (CFC) performance curves were regenerated to accommodate the change in the service water temperature. A CFC performance curve at 97.3 °F is used for the first hour of the accident; the CFC performance curve for a service water temperature of 107 °F is used for the rest of the accident. The CFC delay time was increased from 92 seconds to 115 seconds.

Containment Pressure Analysis Results

Case	New Analysis of Record	New Analysis of Record	Old Analysis of Record	Old Analysis of Record	Design Limit
DEPS Peak Pressure	44.86 psig	875.6 sec	43.8 psig	552 sec	54 psig
DEPS Peak Vapor Temperature	264.39 °F	874.6 sec	263 °F	552 sec	280 °F
DEHL Peak Pressure	43.42 psig	18.51 sec	43.6 psig	18.8 sec	54 psig
DEHL Peak Vapor Temperature	263.57 °F	18.23 sec	264 °F	18.7 sec	280 °F

DEPS – Double Ended Pump Suction
DEHL – Double Ended Hot Leg

Plant Impacts of Change

10CFR50, Appendix J Program Review

- Technical Specification 5.5.17, "Containment Leakage Rate Testing Program," describes the calculated peak containment internal pressure during the design basis LOCA, P_a. The containment leak rate testing program uses P_a when leak testing containment, containment isolation valves, and containment penetrations, including the containment airlock, in accordance with 10 CFR Part 50 Appendix J.

  - Procedures that perform the following tests will require revision to reflect the revised calculated peak containment internal pressure:

    - 10 CFR Part 50, Appendix J, Type A testing
    - 10 CFR Part 50, Appendix J, Type B testing
    - 10 CFR Part 50, Appendix J, Type C testing

Containment Leakage Review

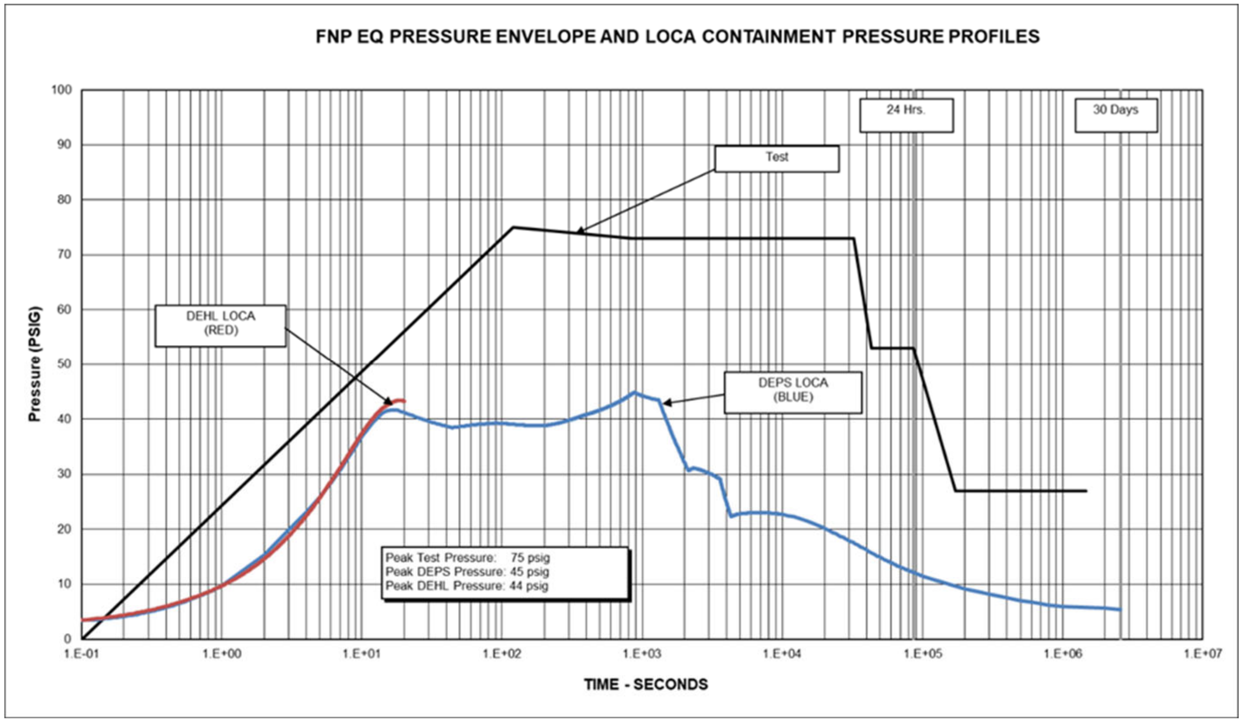
- The change in P_a does not affect the offsite radiological consequences of a LOCA as previously analyzed in the Farley Updated Final Safety Analysis Report Section 15.4.1.7.
  - The LOCA offsite radiological dose consequence analysis is based on the maximum allowable containment leakage rate of 0.15% of containment air weight per day
  - Since the maximum allowable containment leakage rate is not being revised, containment leakage assumed in the LOCA analysis is not impacted

- The change in P_a does not affect the analysis of radiological consequences of a LOCA with respect to radiological dose to the control room operators.
  - Calculated control room operator dose during a LOCA is dependent on the maximum allowable containment atmosphere leakage rate and is unaffected by the change in calculated peak containment internal pressure

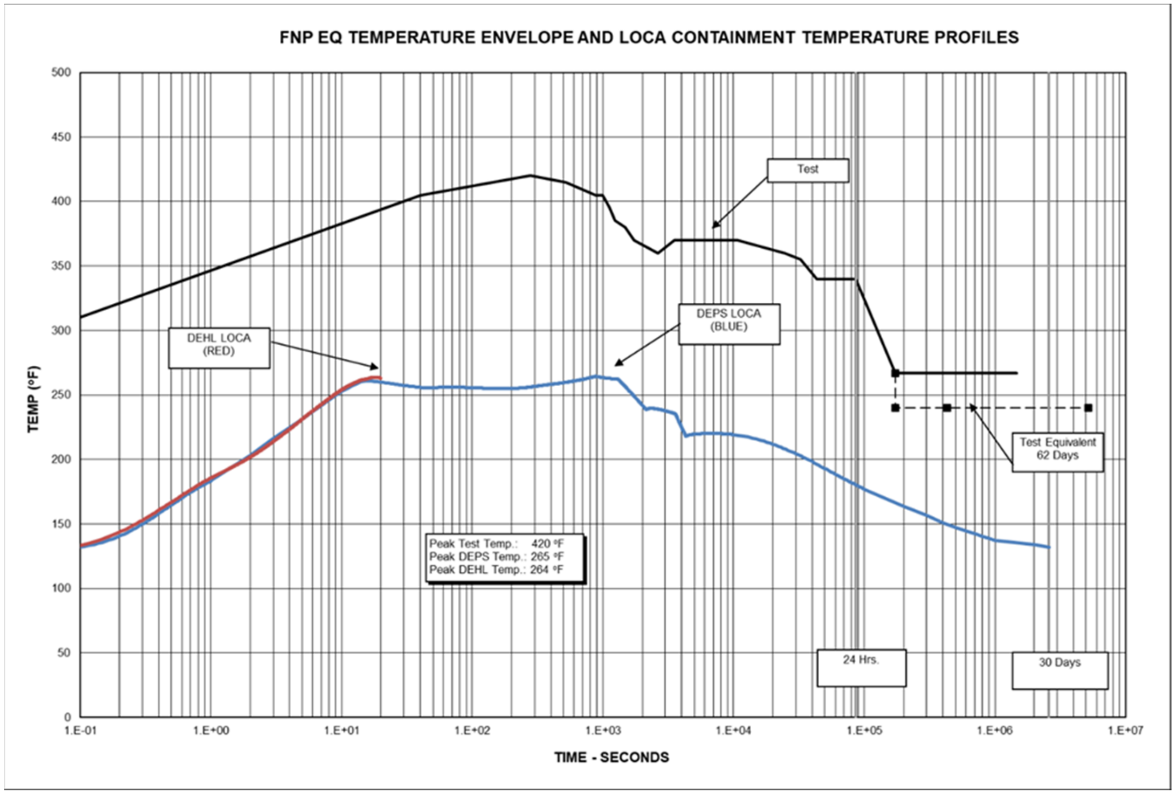
Environmental Qualification of Equipment

- The change in P_a does not affect environmentally qualified equipment within containment. As shown in the following figures, the containment pressure and temperature using the corrected LOCA M&E releases remain within the bounding containment pressure and temperature profile used to qualify equipment.

Environmental Qualification of Equipment - Pressure



Environmental Qualification of Equipment - Temperature



Precedent

Precedents

- Letter from V. C. Summer Nuclear Station to NRC, dated October 6, 2017, License Amendment Request LAR-14-01541 Integrated Leak Rate Test Peak Calculated Containment Internal Pressure Change (ML17279A715)
- Approved by Letter from NRC to V. C. Summer Nuclear Station, dated June 28, 2018, Issuance of Amendment re: Integrated Leak Rate Test Peak Calculated Containment Internal Pressure Change (ML18141A668)

- Letter from Millstone Power Station Unit 3 to NRC, dated April 25, 2013, License Amendment Request to Revise Technical Specification 6.8.4.F for Peak Calculated Containment Internal Pressure (ML13120A158)
- Approved by Letter from NRC to Millstone Power Station Unit 3, dated April 8, 2014, Issuance of Amendment Regarding Calculated Containment Internal Pressure (ML14073A055)

Questions?