Southern Nuclear

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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant – Units 1 and 2
License Amendment Request to Revise Technical Specification 5.5.17, "Containment Leakage Rate Testing Program" to Increase Calculated Peak Containment Pressure

Pursuant to the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (CFR), Southern Nuclear Operating Company (SNC) hereby requests a license amendment to Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License NPF-2 and Unit 2 Renewed Facility Operating License NPF-8. The proposed amendment revises the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA) described in Technical Specifications (TS) 5.5.17, "Containment Leakage Rate Testing Program." The peak calculated containment internal pressure, P_a, is increased from 43.8 psig to 45 psig. The increased P_a remains below the containment design limit of 54 psig.

The enclosure provides a basis for the proposed change. Attachments 1 and 2 contain marked-up TS pages and revised TS pages, respectively.

SNC requests approval of the proposed amendment by December 13, 2022. The proposed changes would be implemented within 60 days after issuance of the amendment.

In accordance with 10 CFR 50.91, a copy of this application, including attachments, is being provided to the designated Alabama Official.

This letter contains no regulatory commitments. If you have any questions, please contact Ryan Joyce at 205.992.6468.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of December 2021.

Respectfully submitted,

C. A. Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/was/cbg

Enclosure: Basis for Proposed Changes

Attachments:
1. Proposed Technical Specification Changes (Marked-up Pages)
2. Revised Technical Specification Pages

cc: Regional Administrator, Region II
NRR Project Manager – Farley 1 & 2
Senior Resident Inspector – Farley 1 & 2
Alabama – State Health Officer for the Department of Public Health
RType: CFA04.054

ENCLOSURE

Farley Nuclear Plant – Units 1 and 2

Revise Technical Specification 5.5.17, "Containment Leakage Rate Testing Program" to Increase Calculated Peak Containment Pressure
Basis for Proposed Changes

Basis for Proposed Changes

1.0 SUMMARY DESCRIPTION

Southern Nuclear Operating Company (SNC) requests an amendment to Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License NPF-2 and Unit 2 Renewed Facility Operating License NPF-8. The proposed amendment revises the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA) described in Technical Specifications (TS) 5.5.17, “Containment Leakage Rate Testing Program.” The peak calculated containment internal pressure, P_a, is increased from 43.8 psig to 45 psig.

2.0 DETAILED DESCRIPTION

2.1 Existing System Design

The containment is a prestressed, reinforced concrete cylindrical structure with a shallow domed roof and a reinforced concrete foundation slab. A 1/4-in.-thick welded steel liner is attached to the inside face of the concrete. The floor liner is installed on top of the foundation slab and is then covered with concrete.

The containment completely encloses the reactor, the reactor coolant systems, the steam generators, and portions of the auxiliary and engineered safeguards systems. It ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the reactor coolant system occurs. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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 and the design temperature limit is 280 °F. The containment system functional design meets the NRC acceptance criteria contained in General Design Criteria 16, 38 and 50 of 10 CFR Part 50.

2.2 Current Technical Specification Requirements

Current TS 5.5.17, “Containment Leakage Rate Testing Program”

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

2.3 Reason for the Proposed Change

Westinghouse Nuclear Safety Advisory Letters (NSALs) identified modeling and material property errors in the analyses for M&E releases. The affected calculations were updated, and it was determined that the errors affected the predicted containment responses. The proposed change reflects updated calculations of the LOCA M&E releases and corresponding containment responses.

2.4 Description of the Proposed Change

Proposed TS 5.5.17, “Containment Leakage Rate Testing Program”

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

3.0 TECHNICAL EVALUATION

The P_a change is justified by evaluating the elements of the containment response analyses that predict the maximum containment pressure during a Loss of Coolant Accident (LOCA). These elements are described below. The LOCA M&E analyses were revised to correct errors documented in three Westinghouse NSALs.

3.1 WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design"

WCAP-10325-P-A (Reference 1) describes the Nuclear Regulatory Commission (NRC) approved method for computing M&E releases into containment by a LOCA. The method uses a suite of computer codes to model the initial blowdown, refill, reflood, and long-term reflood mass and energy releases in accordance with 10 CFR 50 Appendix K, “ECCS Evaluation Models.” Decay heat and stored energy are considered in the method, which is further supported by sensitivity studies on break location, condensation, and other phenomena. SNC has revised the LOCA M&E release calculation using the method in the current licensing basis. However, the inputs have been revised to correct the modeling and material property errors identified in the NSALs discussed below. Additional details on the application of the method to support this license amendment request are provided in Section 3.3.

3.2 Nuclear Safety Advisory Letters

From 2006 to 2014, several Westinghouse NSALs affected LOCA M&E release analyses, and consequently, the computed containment pressure-temperature responses. The extent to which each NSAL affected FNP is described below.

• NSAL 14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," (Reference 2) notified licensees that the mass and energy releases were computed, "...using the density for stainless steel in determining the mass of the SG tubes and the specific heat (C_p) of stainless steel for the stored metal energy. Since all current Westinghouse-designed SGs use either Alloy 600 or Alloy 690 material for the SG tubes, there is a deviation from as built plant parameters..." The FNP Final Safety Analysis Report (FSAR), Section 5.5.2 states that the steam generators are fabricated from INCONEL® 690 material. Therefore, NSAL 14-2 affected FNP.

• NSAL 11-5, "LOCA Mass and Energy Release Calculation Issues" (Reference 3)
Six modeling issues comprise NSAL 11-5:

1. The reactor vessel modeling did not include all the appropriate vessel metal mass available from the component drawings.
2. The reactor vessel modeling did not include all the appropriate vessel metal mass in the reactor vessel barrel/baffle region.

3. The reactor coolant pump (RCP) homologous curve input incorrectly included an absolute zero-point coordinate.
4. The RCP homologous curve input incorrectly contained a sign error in a coordinate value.
5. The LOCA M&E release analysis initializes at a non-conservative (low) steam generator (SG) secondary pressure condition.
6. An error was found in the EPITOME computer code (WCAP-10325-P-A methodology only) that is used to determine the M&E release rate during the long-term (i.e., post-reflood) SG depressurization phase of the LOCA transient.

According to Table 1 of NSAL 11-5, issues 1, 2, 5, and 6 affected the FNP M&E calculations.

• NSAL 06-6, "LOCA Mass and Energy Release Analysis" (Reference 4)
Eight modeling issues comprise NSAL 06-6:

1. Area of the Downcomer in the REFLOOD Code
2. Area of the Upper Plenum in the FROTH Code
3. Review of Other FROTH Inputs
4. WCAP-10325-P-A Model Features
5. Main Feedwater Addition Following a Reactor Trip
6. AFW System Purge and Unisolatable Volumes
7. AFW Flow for FROTH Code
8. Possibility of Asymmetric AFW Flow

Of these issues, only issues 5 and 6 were found to affect FNP M&E analyses.

3.3 LOCA Mass and Energy Release Analysis

To address the issues identified in Section 3.2, FNP-specific M&E release analyses were performed in accordance with the NRC approved WCAP-10325-P-A methodology. These analyses are performed in the same manner as the current analysis of record (AOR) described in the FNP FSAR Section 6.2, as described below.

In the event of a hypothetical LOCA the release of the coolant from the rupture area will cause the high pressure, high temperature fluid to rapidly flash to steam and water within the containment. The release of this M&E will result in a rise in the pressure and temperature of the containment atmosphere. The rate and magnitude of the pressure increase depend upon the nature, location, and size of the rupture. In order to establish the controlling rupture, a spectrum of primary and secondary coolant breaks is considered. The reactor coolant breaks examined are from a condition of full rated power.

With respect to break size, the double-ended guillotine break has been found to be limiting due to larger mass flowrates during the blowdown phase of the transient. During the reflood and froth phases, break size has little effect on the releases. Three distinct locations in the reactor coolant system (RCS) loop are postulated for pipe rupture for any release:

- Hot leg (between vessel and steam generator)
- Cold leg discharge (between pump and vessel)
- Pump suction (between steam generator and pump)

Of these locations, the Double-Ended Hot Leg (DEHL) and Double-Ended Pump Suction (DEPS) have generated the most limiting short- and long-term peak containment conditions, respectively. The DEPS case assumes minimum safeguards (safety injection - SI) flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The M&E release for these cases have been reanalyzed for this

license amendment request. Since the cold leg discharge break is non-limiting, it has not been reanalyzed.

The M&E releases are determined using the evaluation model described in WCAP-10325-P-A which relies on the following computer codes:

- SATAN-VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, M&E flowrates, and energy transfer between primary and secondary systems as a function of time.
- WREFLOOD addresses the portion of the LOCA transient where the core reflood phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break, and when water supplied by the emergency core cooling system (ECCS) refills the reactor vessel and provides cooling to the core.
- FROTH models the post-reflood portion of the transient. FROTH is used for the steam generator heat addition calculation from the broken and intact loop steam generators.
- EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient.

The American Nuclear Society (ANS) Standard 5.1, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979, was used in the LOCA M&E release model for the determination of decay heat energy. Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E releases analysis include the following:

- Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- Fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS Standard 5.1-1979.
- The fuel has been assumed to be at full power for 10^8 seconds.
- The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
- Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

The following assumptions were employed to ensure that the M&E releases are conservatively calculated, thereby maximizing energy release to containment. These assumptions are unchanged from the current AOR.

1. Maximum expected operating temperature of the RCS (100% full-power conditions).
2. Allowance for RCS temperature uncertainty (+6.0°F).
3. Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty).
4. Analyzed core power of 2830.5 MWt.
5. Conservative heat transfer coefficients (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer).
6. Allowance in core stored energy for effect of fuel densification.
7. A margin in core stored energy to bound Optimized Fuel Assembly (OFA) fuel (3.75 full power seconds (FPS)).
8. An allowance for RCS initial pressure uncertainty (+50 psi).
9. A maximum containment backpressure equal to design pressure (54 psig).

10. Allowance for RCS flow uncertainty (-2.4%).
11. Steam generator tube plugging leveling (0% uniform).
    - Maximizes reactor coolant volume and fluid release.
    - Maximizes heat transfer area across the SG tubes.
    - Reduces coolant loop resistance, which reduces the dP upstream of the break for the pump suction breaks and increases break flow.

The following inputs were updated for the M&E release calculation revision:

1. The codes used in the M&E release methodology, owned by Westinghouse, were changed to correct the errors discussed in the NSALs. Most of those changes are input changes, however, there are changes to the method of evaluation. These changes are:
    - Treatment of inactive metal mass
    - Turning on the drift flux model in the SATAN-VI 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.
2. 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. This brought the RWST temperature input for M&E release into agreement with the value used in the containment response analysis, and in agreement with plant design inputs. This input change removes SNC controlled analytical margin.

Using the NRC approved WCAP-10325-P-A methodology, M&E releases were calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. The DEPS case considers the minimum SI. For the DEHL case, the releases were calculated only for the blowdown. The data were then input to the containment analysis to determine the peak containment pressure.

3.4 LOCA Containment Analysis

The containment pressure analyses to determine the limiting LOCA were performed using the GOTHIC computer program (version 8.1) that was developed for the purpose of transient analysis of atmospheres in multicompartment containments of water-cooled nuclear power plants. The use of GOTHIC for the FNP containment pressure analyses was reviewed and approved by the NRC as described in the FNP FSAR Section 6.2. The GOTHIC code used to calculate the containment pressure/temperature response was upgraded from version 6.0 to version 8.1. The change in code version does not result in a numerically significant departure from the previous analysis and does not yield a benefit to peak pressure and temperature results.

The DEHL break and the minimum SI DEPS break are the analyzed scenarios. The DEHL results in the higher peak pressure that occurs in the initial blowdown, while the DEPS minimum SI case results in the overall limiting condition (which occurs in the long-term).

The initial conditions for this analysis are:

- Pressure = 17.7 psia (3 psig, which is the TS 3.6.4 maximum pressure)
- Vapor Temperature = 127 °F
- Liquid Temperature = 127 °F
- Relative Humidity = 50 %

- Liquid Volume Fraction = 4.0E-07

The liquid volume fraction is set to an arbitrary low fraction to ensure there is some liquid in the sump when the accident is initiated. These initial conditions are consistent with the existing AOR.

The following design inputs were updated during the calculation revision to reflect the current state of the plant:

1. The post-LOCA service water pond temperature calculation is revised, resulting in a higher peak service water temperature during the 30-day post-LOCA transient. 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. This input change is conservative.
2. The containment fan cooler (CFC) performance curves were regenerated to accommodate the change in the service water temperature. The previous performance curve was based on a service water temperature of 97.3 °F. A new analysis of the containment fan cooler was performed which re-generated the CFC performance curve at a service water temperature of 97.3 °F, and created a CFC performance curve at a service water temperature of 107 °F. The performance curve at 97.3 °F is used for the first hour of the accident; afterward, the performance curve at 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. These input changes are conservative.

All input changes made to the analysis were evaluated under SNC's 50.59 process and found to not need prior NRC approval. No changes to the model (e.g., heat sinks) were made in this analysis revision.

3.4.1 LOCA Containment Analysis Results

Using the method described in FSAR Section 6.2.1.3.4.2, the resulting containment pressure responses for the updated M&E releases at the DEHL and DEPS locations were developed. The peak temperature and pressure for the DEPS minimum ESF case and the DEHL case are reported below for both the existing AOR and the newly calculated AOR. Graphs of the DEPS containment pressure and vapor temperature are presented below in Figures 1 and 2. Graphs of the DEHL containment pressure and vapor temperature are presented in Figures 3 and 4.

Case	New AOR	New AOR	Old AOR	Old AOR	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

The effect of the revised design inputs are as follows:

1. Updated M&E Release: The incorporation of the revised NSAL M&E releases lead to an increase in the calculated peak pressure and temperature. However, the increases in pressure and temperature remain below the design limits of the containment system, with margin to those limits. Therefore, the impact on the containment fission product barrier of incorporating the NSALS is acceptable.

2. Increase in Service Water Temperature: The residual heat removal/containment cooling water heat exchanger does not come into service until switchover to recirculation. Switchover occurs after the peak pressure and peak temperature are calculated to occur. This analytical change did not impact the peak pressure or temperature.
3. Containment Fan Cooler Performance Curves: A comparable CFC heat removal curve was used for the first hour of the analysis, which includes the time at which peak pressure and peak temperature are calculated. After the first hour, a CFC heat removal curve based on a higher service water temperature is used. Because the performance of the CFC fan cooler is based on the original service water temperature from the previous analysis for the first hour, there is a negligible effect on the calculated peak pressure and temperature. This analytical change did not impact the peak pressure or temperature.

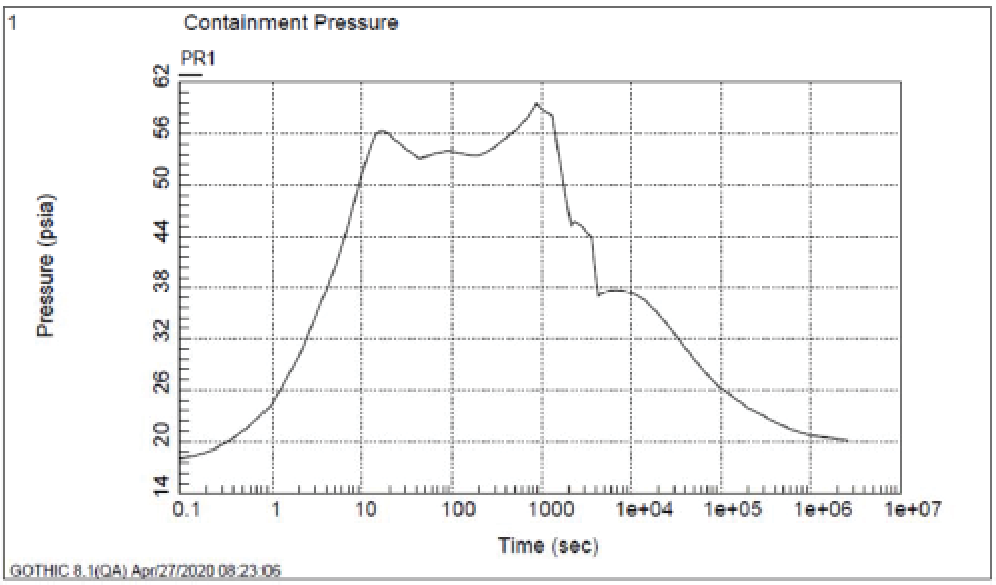

Figure 1: Containment Pressure for the DEPS Minimum ESF Case
* Note: Units in psia.

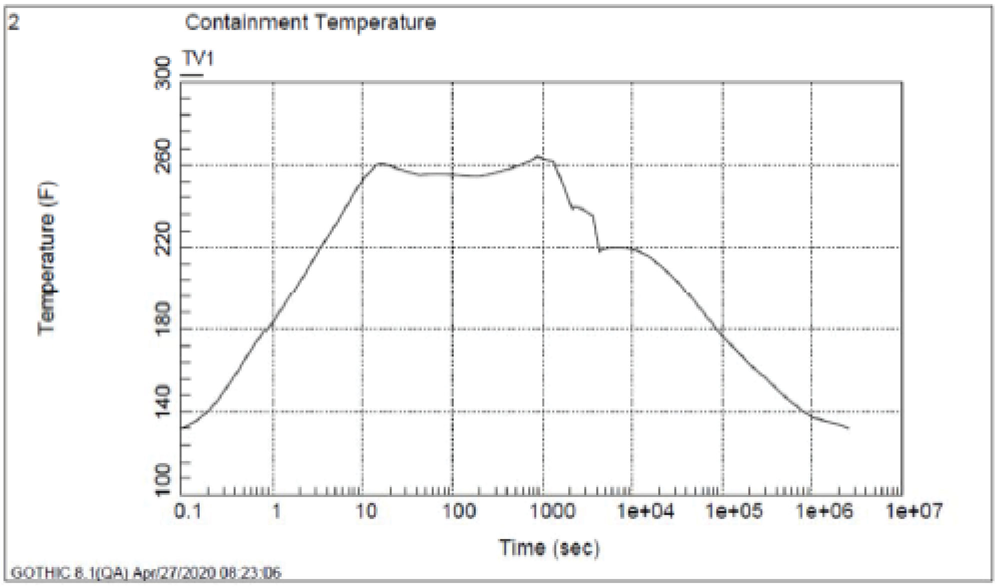

Figure 2: Containment Temperature for the DEPS Minimum ESF Case

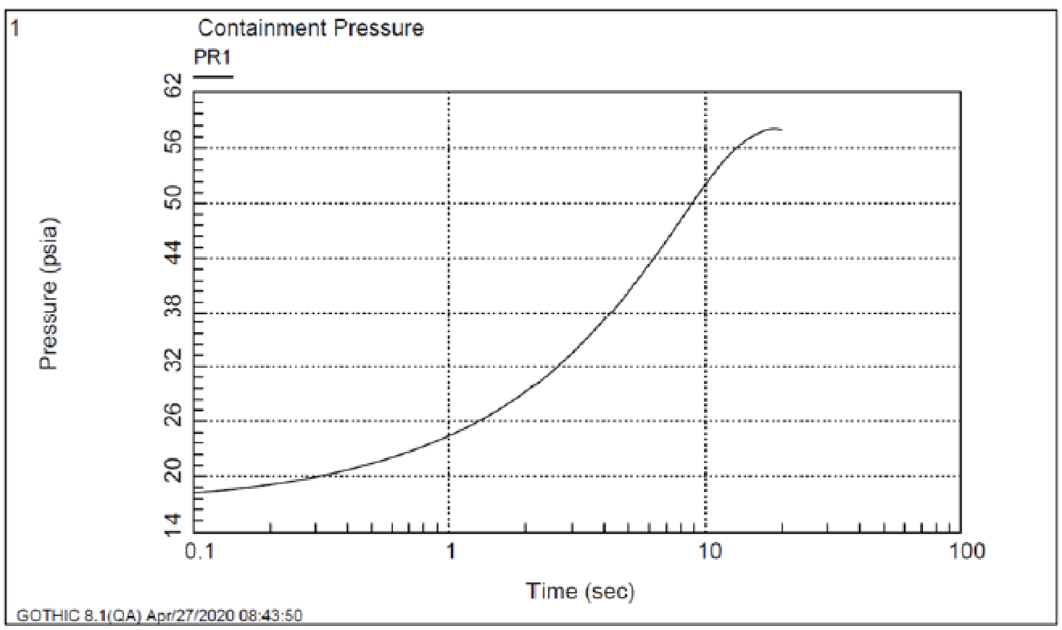

Figure 3: Containment Pressure for the DEHL Case
* Note: Units in psia

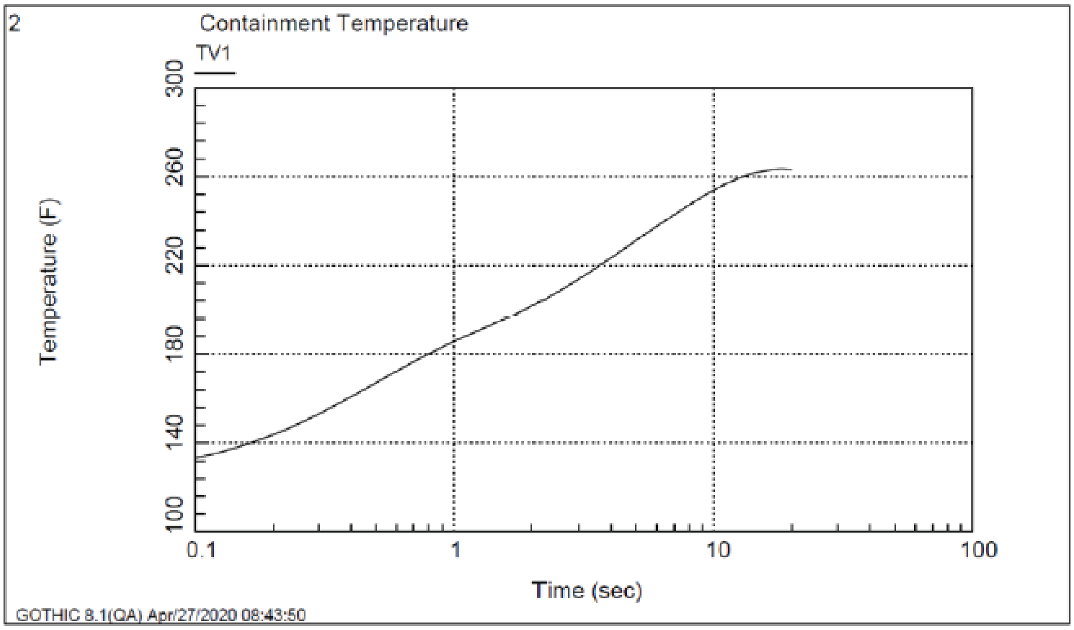

Figure 4: Containment Temperature for the DEHL Case

As shown in the results above, the maximum containment pressure is slightly less than 45 psig based on the DEPS case. Therefore, a change in P_a to 45 psig is justified by the analysis results.

3.5 Plant Impacts of Change in Containment Pressure

<u>10CFR50, Appendix J Program Review</u>
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.

Upon NRC approval, this increase in P_a would be reflected in the 10 CFR Part 50, Appendix J containment leak rate testing procedures. The procedures that govern 10 CFR Part 50, Appendix J, Type A testing will be changed to reflect the revised calculated peak containment internal pressure in Technical Specification 5.5.17. Title 10 CFR Part 50, Appendix J, Type A testing is required by ANS/ANSI 56.8-2002 to be performed at a pressure not less than 0.96 times the calculated peak containment pressure (P_a) and not more than the containment design pressure. These values will be updated in the procedure prior to the next performance of the Type A test for either unit.

10 CFR Part 50, Appendix J, Type B and C test procedures also require revision upon approval of the proposed license amendment. The applicable ANS/ANSI Standard (56.8-2002) requires that Type B and C testing be performed at a pressure not less than P_a (except for airlock door seals, which may have a lower pressure specified) and not more than 1.1 times P_a when a higher differential pressure results in increased sealing. Site procedures for Type B and C testing require that testing be performed within a range of pressures that, with the revised P_a, will need to be updated to address the change to the minimum pressure allowed, P_a.

<u>Dose Consequences Review</u>

The change in P_a does not affect the offsite radiological consequences of a LOCA as previously analyzed in the FNP FSAR 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. The analysis assumes that containment leakage during the first 24 hours of the event is 0.15% of containment atmosphere by weight and 0.075% of containment atmosphere by weight afterward. It also assumes that the release of radionuclides to the containment is instantaneously mixed with containment air within the containment free air volume. Since the maximum allowable containment leakage rate is not being revised, containment leakage assumed in the LOCA analysis is not impacted. Therefore, the increase in the calculated peak containment internal pressure does not impact the offsite radiological consequences of the LOCA accident analysis.

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, as discussed above. Since the maximum allowable containment leakage rate is not being revised, dose to the control room operators is not affected by a change in peak containment pressure.

The post-accident sump temperature profile is changed as a consequence of the service water temperature input change. The sump temperature profile is an input into the RWST back-leakage dose contribution model for the LOCA accident. The impact of the change in the sump water temperature profile was evaluated in a revision to the LOCA dose consequence analysis. The revised results were negligibly increased due to higher sump water temperatures later in the accident. The increase in results was within the round-off of reported results. Therefore, the increase in sump temperature as a result of the service water input change does not impact the offsite consequences of the LOCA accident analysis.

<u>Equipment Environmental Qualification Review</u>

A review of the effects of this change in M&E release on the environmental qualification (EQ) of equipment in containment was performed. Figures 5 and 6 present the pressure and temperature results of the DEPS and DEHL cases as compared to the EQ limits for equipment inside containment.

The change in P_a does not affect environmentally qualified equipment within containment. As shown in Figure 5, the containment pressures using the corrected LOCA M&E releases remain within the bounding containment pressure profile used to qualify equipment. Therefore, an increase in peak calculated containment internal pressure to 45 psig does not affect the EQ of equipment within containment.

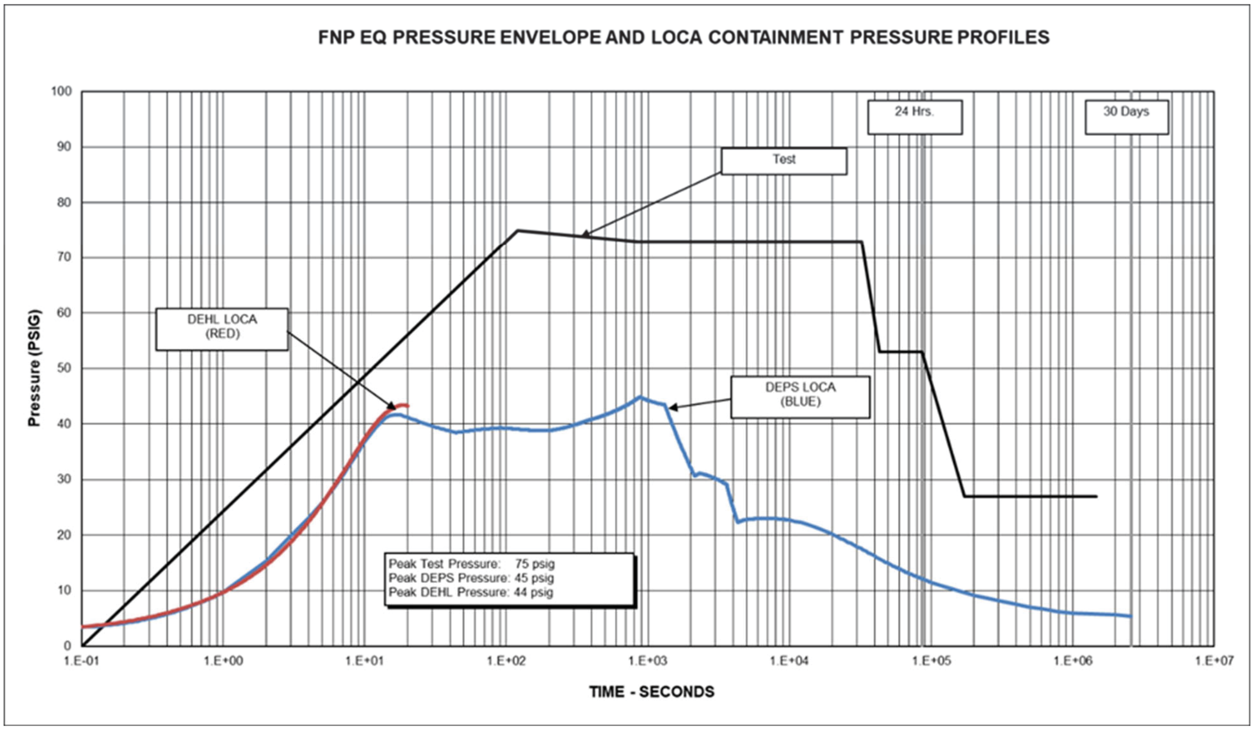

Figure 5: Environmental Qualification and LOCA Containment Pressure Profiles

The containment temperatures using the corrected LOCA M&E releases remain within the bounding containment temperature profile used to qualify equipment as shown in Figure 6. Therefore, the post-accident operating time of the environmentally qualified equipment is unaffected.

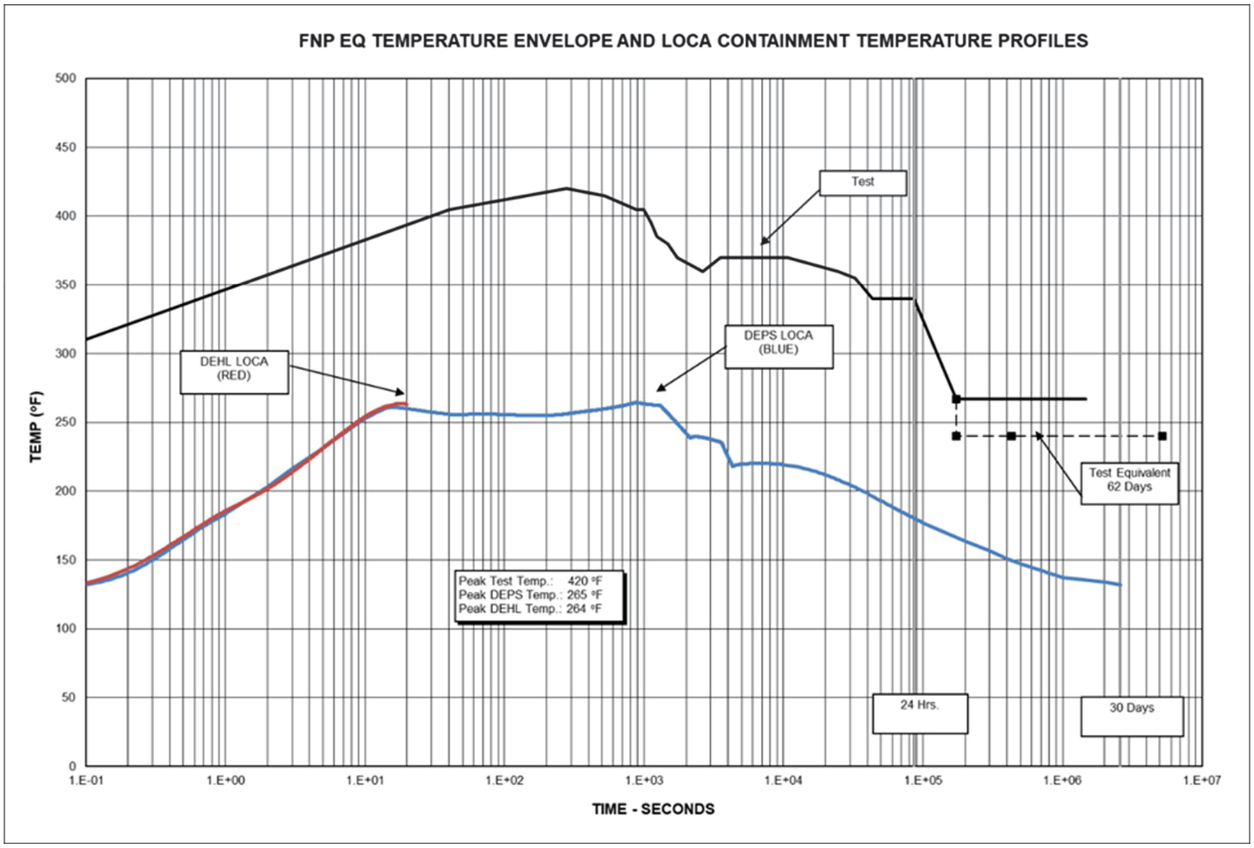

Figure 6: Environmental Qualification and LOCA Containment Temperature Profiles

Temperature and pressure margins in the FNP EQ program are applied in accordance with 10 CFR 50.49 and Regulatory Guide 1.89. These margins are in addition to the EQ bounding temperature and pressure profiles for equipment qualified to 10 CFR 50.49. With the margin applied, the calculated temperature and pressure profiles remain bounded by the FNP EQ equipment test profiles. EQ Program documents have been revised based on these new calculated pressure and temperature profiles with no negative impacts.

The proposed change in calculated peak containment internal pressure for the design basis loss of coolant accident does not result in any impact on any area of the plant outside containment. As a result, environmental conditions outside containment remain bounded by existing site calculations.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

Criterion 16 - Containment Design
Reactor containment and associated systems are provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the

containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Conformance to Criterion 16
The containment is designed to provide protection for the public from consequences in the unlikely event of the LOCA, which is based on a postulated break of the reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe. The containment is designed to safely sustain all internal and external loading conditions resulting from postulated transients and accidents. Functional capability of the containment will be maintained for as long as required to protect the public. Due consideration is given to site factors and local environments as they relate to public health and safety. To satisfy this GDC, it is verified that the peak calculated pressure is within the containment design pressure of 54 psig (68.7 psia) for all containment pressurization events.

Criterion 38 - Containment Heat Removal
A system to remove heat from the reactor containment is provided. The system safety function is to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities are provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Conformance to Criterion 38
Three systems are provided to reduce containment atmosphere temperature and pressure and/or to remove heat from the containment under post-accident conditions. These are the low head safety injection/residual heat removal system, the containment spray system, and the containment cooling system. The containment heat removal systems are designed to ensure that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the systems from accomplishing their design safety functions. To satisfy this GDC, it is shown that with one train of containment safeguards equipment (assuming single failure of the other train) that the temperature and pressure following a LOCA is rapidly reduced to acceptably low values and that it is maintained low for at least 30 days.

Criterion 50 - Containment Design Basis
The reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin reflects consideration of the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning; the limited experience and experimental data available for defining accident phenomena and containment responses; and the conservatism of the calculational model and input parameters.

Conformance to Criterion 50
The design of the containment is based on the LOCA, coupled with the partial loss of the redundant engineered safety features, which produces the post-LOCA temperature and

pressure conditions described in FNP FSAR section 6.2.1. A minimum safety margin of 10 percent between the peak calculated post-LOCA containment pressure and the design pressure has been provided. To satisfy this GDC, conservative inputs are used in the GOTHIC analytical model and it is verified that the available engineered safety features equipment is capable of bringing and maintaining the containment temperature and pressure to acceptably low levels. The 10 % margin between the peak calculated containment pressure and the design pressure of 54 psig is maintained.

These GDC continue to be met by the change in calculated peak containment pressure.

4.2 Precedent

- 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 (Reference 5)
- 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 (Reference 6)

This LAR requested three changes to the V. C. Summer TS. Two of the changes address the regulatory requirements for containment leak rate testing. The third change is an increase in P_a, based on the same NSAL changes as the one requested here. Note that the V. C. Summer request also included different plant-specific changes than are described above for FNP.

- 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 (Reference 7)
- Letter from NRC to Millstone Power Station Unit 3, dated April 8, 2014, Issuance of Amendment Regarding Calculated Containment Internal Pressure (Reference 8)

The LAR requested a change in P_a based on a change in the M&E release due to errors identified in NSAL 11-05 and one additional error. The LAR addressed the effects of a change in P_a on plant programs.

4.3 No Significant Hazard Consideration Determination Analysis

Southern Nuclear Operating Company (SNC) has evaluated the proposed changes to the Technical Specifications (TS) using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

The proposed amendment revises the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA) described in TS 5.5.17, “Containment Leakage Rate Testing Program.” The peak calculated containment internal pressure, P_a, is increased from 43.8 psig to 45 psig. This increase in P_a is due to an increase in the calculated mass and energy released into containment during the blowdown phase of the design basis LOCA event.

As required by 10 CFR 50.91(a), the SNC analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No
The activity does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents analyzed in the Final Safety Analysis Report. As such, the containment itself and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. The updated P_a value reflects the updated mass and energy release and containment response calculations, ensuring a sound technical basis for the local and integrated leakage tests. Radiological consequences will continue to be evaluated at the TS allowed leakage, L_a, of 0.15 percent by weight of air, which will not be increased despite the increase in P_a.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No
The proposed changes do not require any change in the operation of the plant or a change in the methods governing normal plant operation. The requested change to TS 5.5.17 is consistent with the safety analysis assumptions discussed in this license amendment request. The containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No
The proposed TS change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. Using the same Nuclear Regulatory Commission approved analysis methodology as previously used, the updated mass and energy release and containment response analyses corrected input errors identified by Westinghouse, the methodology owner. The correction of these errors resulted in a slightly higher calculated containment peak pressure and temperature than that of the current licensing basis but does not pose a significant challenge to the containment pressure or temperature design limit.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

4.4 Conclusion

In conclusion, based on the considerations discussed above, SNC concludes: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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 the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design," dated March 1979
2. NSAL 14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," dated March 31, 2014
3. NSAL 11-5, "LOCA Mass and Energy Release Calculation Issues," dated July 25, 2011
4. NSAL 06-6, "LOCA Mass and Energy Release Analysis," dated June 6, 2006
5. Letter from Virgil C. Summer Nuclear Station (VCSNS) to the NRC, License Amendment Request LAR-14-01541 Integrated Leak Rate Test Peak Calculated Containment Internal Pressure Change, dated October 6, 2017 (ML17279A715)
6. Letter from NRC to Virgil C. Summer Nuclear Station, Unit No. 1 - Issuance of Amendment Re: Integrated Leak Rate Test Peak Calculated Containment Internal Pressure Change, dated June 28, 2018 (ML18141A668)
7. 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)
8. Letter from NRC to Millstone Power Station Unit 3, dated April 8, 2014, Issuance of Amendment Regarding Calculated Containment Internal Pressure (ML14073A055)

Farley Nuclear Plant – Units 1 and 2

Revise Technical Specification 5.5.17, “Containment Leakage Rate Testing Program” to Increase Calculated Peak Containment Pressure

Attachment 1

Proposed Technical Specification Changes (Marked-up Pages)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~43.8~~45 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq$ 1.0 L_a. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the combined Type B and C tests, and $\leq$ 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq$ 0.05 L_a.

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Farley Nuclear Plant – Units 1 and 2
Revise Technical Specification 5.5.17, “Containment Leakage Rate Testing Program” to Increase Calculated Peak Containment Pressure

Attachment 2

Revised Technical Specification Pages

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

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

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

    1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

    3. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.

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