Robinson / NRC Pre-submittal Meeting:
LAR to Add Feedwater Isolation on SG High-High & Remove Obsolete Methods

June 7, 2022

Duke Energy Attendees

Ryan Treadway (Manager, Nuclear Fleet Licensing)

Joshua Duc (Nuclear Fleet Licensing)

Jeff Abbott (Manager, Fleet Nuclear Fuels Engineering)

Christy Ray (Fleet Nuclear Fuels Engineering)

Scott Jackson (RNP Engineering)

Brad Hearne (RNP Engineering)

Fred Lane (RNP Operations)

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Agenda

- Current and Proposed Technical Specification (TS) / Surveillance Requirement (SR)
  - Steam generator overfill protection: add feedwater isolation on SG level high-high to TS 3.3.2
  - Revise TS 2.1.1.1 and TS 5.6.5.b to reflect obsolete analytical methods
- System Design and Operation
- Reason for the Proposed Change
- Justification
- Precedent
- Schedule

Current and Proposed TS / SR

No changes, included for reference only

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

D. One channel inoperable	------------------NOTE------------------ For Function 4.c, a channel may be taken out of the trip condition for 6 hours for maintenance. ---	
	D.1 Place channel in trip.	6 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	18 hours

Current and Proposed TS / SR

Table 3.3.2-1 (page 4 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2(f),3(f)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. ESFAS Interlocks						
a. Pressurizer Pressure Low	1,2,3	3	H	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 2005.11 psig	2000 psig
b. T_{avg} – Low	1,2,3	1 per loop	H	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 544.50 °F	543°F

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.
(f) Except when all MFIVs, MFRVs, and bypass valves are closed or isolated by a closed manual valve.

c. SG Water Level - High-High	1,2(f),3(f)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤76.16%	75%

Note:
SR 3.3.2.1 = CHANNEL CHECK
SR 3.3.2.4 = CHANNEL OPERATIONAL TEST (COT)
SR 3.3.2.7 = CHANNEL CALIBRATION

Current and Proposed TS / SR

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest cold leg temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP correlation ~~and ≥ 1.17 for the XNB correlation~~.

2.1.1.2 The peak fuel centerline temperature shall be maintained < [4901 - (1.37 x 10^{-3} x (Burnup, MWD/MTU))] °F.

Current and Proposed TS / SR

TS 5.6.5, "Core Operating Limits Report (COLR)"

Replace text in items 2, 3, and 8 with "Deleted"

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The approved version shall be identified in the COLR. These methods are those specifically described in the following documents:

1. Deleted

2. ~~XN-NF-84-73(P), "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," approved version as specified in the COLR.~~

3. ~~XN-NF-82-21(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.~~

4. Deleted

5. XN-75-32(A), "Computational Procedure for Evaluating Rod Bow," approved version as specified in the COLR.

6. Deleted

7. Deleted

8. ~~XN-NF-78-44(A), "Generic Control Rod Ejection Analysis," approved version as specified in the COLR.~~

(continued)

Current and Proposed TS / SR

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements (continued)

9. ~~XN-NF-621(A), "XNB Critical Heat Flux Correlation," approved version as specified in the COLR.~~

Replace text in items 9 and 11 with "Deleted"

10. Deleted

11. ~~XN-NF-82-06(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.~~

12. Deleted

13. Deleted

Current and Proposed TS / SR

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. Deleted
15. Deleted
16. ~~ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.~~
17. ~~ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," approved version as specified in the COLR.~~
18. ~~ANF-89-151(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.~~
19. ~~EMF-92-081(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.~~
20. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.
21. ~~XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.~~
22. ~~EMF-96-029(P)(A), "Reactor Analysis System for PWRs," approved version as specified in the COLR.~~
23. ~~EMF-92-116, "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.~~
24. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.

Replace text in items 16, 17, 18, 19, 21, 22, and 23 with "Deleted"

Current and Proposed TS / SR

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Replace text in item 25 with "Deleted"

25. ~~EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.~~

26. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods," approved version as specified in the COLR.

27. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," approved version as specified in the COLR.

28. DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," approved version as specified in the COLR.

29. DPC-NE-1008-P-A, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," as approved by NRC Safety Evaluation dated May 18, 2017.

30. DPC-NF-2010-A, "Nuclear Physics Methodology for Reload Design," as approved by NRC Safety Evaluation dated May 18, 2017.

31. DPC-NE-2011-P-A, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" as approved by NRC Safety Evaluation dated May 18, 2017.

32. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated April 10, 2018.

33. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated April 10, 2018.

34. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," approved version as specified in the COLR.

System Design and Operation

- Robinson (RNP) is Westinghouse-designed 3-loop pressurized water reactor (PWR)

- Main Feedwater (MFW) system supplies water to Steam Generators (SGs)
  - Main flow path: Main Feedwater Isolation Valve (MFIV) and Main Feedwater Regulation Valve (MFRV)
  - Bypass flow path: two bypass valves

- FW Isolation on SG Level High-High at 75% level (SG overfill protection)
  - Trips main turbine and MFW pumps
  - Closes MFRV and bypass valves on the affected SG
  - 2 out of 3 channel initiating logic that is safety related
  - Not part of Engineering Safety Features Actuation System (ESFAS)

System Design and Operation

- Background
  - Generic Letter (GL) 89-19
    - SG overfill protection design is sufficiently separate from MFW control system
    - TS include periodic verification of operability of SG overfill protection system
    - NRC closeout letter for RNP dated June 24, 1991 (ML14184A839)
  - RNP conversion to standard TS (STS) – safety evaluation dated October 24, 1997 (ML020560172, ML14175A922, ML14175A924)
    - FW isolation on SG high-high not included in new ESFAS TS Table 3.3.2-1
      - RNP STS submittal: "This Function is not classified as an Engineered Safety Feature in the plant design basis and current licensing basis."
  - RNP UFSAR 15.1.2, "Feedwater System Malfunctions that Result in an Increase in Feedwater Flow"
    - Currently not an explicit analysis, bounded by UFSAR 15.1.3 and 15.4.1 events
    - Does not acknowledge that feedwater must be terminated to prevent SG overfill
    - Approved via safety evaluation dated November 7, 1984 (ML020520268)

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Reason for the Proposed Change

- SG overfill can occur within ~5 minutes of event initiation if MFW is not isolated
- Concerns documented in GL 89-19 still valid and could challenge fission product barrier
  - "…(1) the increased dead weight and potential seismic loads placed on the main steam line and its support should the main steam line be flooded;
  - (2) the loads placed on the main steam lines as a result of the potential for rapid collapse of steam voids resulting in water hammer;
  - (3) the potential for secondary safety valves sticking open following discharge of water or two-phase flow;
  - (4) the potential inoperability of the main steam line isolation valves (MSIVs), main turbine stop or bypass valves, feedwater turbine valves, or atmospheric dump valves from the effects of water or two phase flow…"
- Restore compliance with GL 89-19
  - Correct non-conservative TS to ensure UFSAR Chapter 15 Condition II event does not transition to more serious Condition IV event (Condition II acceptance criteria)
- Remove analytical methods no longer in use due to approval of new Duke Energy methods

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Justification

- UFSAR 15.1.2 reevaluated using NRC-approved Duke method DPC-NE-3009-P-A
  - MFRV to one SG failed open, increasing feedwater flow
  - FW isolation upon reaching uncertainty adjusted SG high-high setpoint
    - Turbine trip would also concurrently occur but is conservatively not modeled in this new analysis
  - Significant margin to Departure from Nucleate Boiling (DNB) and Centerline Fuel Melt (CFM) acceptance criteria
  - Peak primary and peak secondary pressures bounded by UFSAR 15.2.2 event

- Allowable value of 76.16% calculated consistent with other ESFAS setpoints
  - Process measurement accuracy, sensor calibration accuracy, sensor and rack drift, the effects of temperature and pressure on the racks and sensors, etc.
  - Below maximum reliable indicated level (MRIL) of Westinghouse NSAL-02-4

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Justification

- Proposed MODES are 1, 2, and 3
  - Except in MODES 2 and 3 when all MFIVs, MFRVs, and associated bypass valves are closed or isolated by a closed manual valve, as feedwater isolation would be satisfied in this condition
  - In MODES 4, 5 and 6, the MFW system and the turbine generator are not in service
  - Consistent with STS NUREG-1431

- Utilize existing Condition D
  - Specific to channel operability and applies to functions with 2 out of 3 logic
  - Existing Completion Times more conservative than STS NUREG-1431 (72 hrs)

Justification

- SRs 3.3.2.1 (Channel Check), SR 3.3.2.4 (COT), SR 3.3.2.7 (Channel Calibration)
  - Frequency controlled by the Surveillance Frequency Control Program
  - Consistent with other existing Table 3.3.2-1 ESFAS channel-based protection functions
  - NUREG-1431 specifies SR 3.3.2.1, SR 3.3.2.5, SR 3.3.2.9, and SR 3.3.2.10
    - Proposed SRs are consistent with SR 3.3.2.1, SR 3.3.2.5, and SR 3.3.2.9
    - SR 3.3.2.10 to verify ESFAS response times is not applicable to RNP, as it was not included in the approved RNP conversion to STS

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Justification

- GL 89-19: sufficient separation between MFW control and overfill protection
  - All 3 level indication channels used for both overfill protection and MFW control (median selector)
  - Signal isolation devices for separation of protection and control
  - Power supplies through different circuits off the same instrument buses
    - Power to instrument buses is reliable source with alternate supply available
  - The 3 channels do not share common routing
    - 2 protection channels remain operable in the event of a fire in MFW control system

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Justification

- Methods approved to allow Duke Energy in-house analysis for RNP
  - DPC-NE-2005-P, “Thermal-Hydraulic Statistical Core Design Methodology,” safety evaluation March 8, 2016 (ML16049A630)
  - DPC-NE-1008-P, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” safety evaluation May 18, 2017 (ML17102A923)
  - DPC-NF-2010, “Nuclear Physics Methodology for Reload Design,” safety evaluation May 18, 2017 (ML17102A923)
  - DPC-NE-2011-P, “Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors,” safety evaluation May 18, 2017 (ML17102A923)
  - DPC-NE-3008-P, “Thermal-Hydraulic Models for Transient Analysis,” safety evaluation April 10, 2018 (ML18060A401)
  - DPC-NE-3009-P, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” safety evaluation April 10, 2018 (ML18060A401)
  - BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” safety evaluation April 29, 2019 (ML18288A139)

- The methods and DNB correlation limit proposed for deletion are no longer planned for use in RNP design/safety analysis
- Administrative change to remove obsolete information

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Precedent

- GL 89-19 required response from licensees in early 1990’s
  - Turkey Point added SG overfill protection, safety evaluation dated April 28, 1994 (ML013380368)
    - Similar 3-loop Westinghouse PWR with 2 out of 3 SG overfill logic
    - Stated in TS and TS Bases that SG overfill is not part of ESFAS

- Remove obsolete methods
  - Harris safety evaluation dated April 8, 2021 (ML21047A470)
    - Based on the same Duke Energy in-house methods
  - Harris safety evaluation dated March 30, 2012 (ML12058A133)
  - RNP safety evaluation dated December 29, 2011 (ML11342A165)

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Schedule

- Submit LAR by June 30, 2022
- Implementation within 120 days of receipt of safety evaluation

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