Nicole L. Flippin
H. B. Robinson Steam Electric Plant Unit 2
Site Vice President

Duke Energy
3581 West Entrance Road
Hartsville, SC 29550

O: 843 951 1701
F: 843 951 1319
Nicole.Flippin@duke-energy.com

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10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

SUBJECT: License Amendment Request to Add Feedwater Isolation on Steam Generator Level High-High to Technical Specification 3.3.2 and Update the List of Analytical Methods Used in the Determination of Core Operating Limits

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) is submitting a request for an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (RNP), Unit No. 2. The proposed amendment would add a new function to TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation" Table 3.3.2-1 for Feedwater Isolation on Steam Generator (SG) level high-high (i.e., SG overfill protection). In addition, proposed revisions to TS 2.1.1.1 and TS 5.6.5.b are included to reflect the removal of analytical methods no longer applicable for the determination of RNP core operating limits.

On June 7, 2022 Duke Energy and Nuclear Regulatory Commission (NRC) staff participated in a pre-submittal meeting regarding the proposed changes to TS 3.3.2, TS 2.1.1.1, and TS 5.6.5.b.

The Enclosure to this letter provides an evaluation of the proposed changes. Attachments 1 and 2 provide the existing TS and TS Bases pages, respectively, marked to show the proposed changes. The marked-up TS Bases pages are provided for information only.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed changes involve no significant hazards consideration. The basis for this determination is included in the Enclosure.

Duke Energy requests approval of the proposed license amendment within one year of completion of the NRC's acceptance review. Once approved, Duke Energy will implement the amendment within 120 days.

This submittal contains no new regulatory commitments.

In accordance with 10 CFR 50.91, Duke Energy is notifying the state of South Carolina of this license amendment request by transmitting a copy of this letter to the designated state officials. Should you have any questions concerning this letter, or require additional information, please contact Ryan Treadway, Manager – Nuclear Fleet Licensing, at 980-373-5873.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 21, 2022.

Sincerely,

Nicole Flippin

Nicole L. Flippin
Site Vice President

Enclosure:
Evaluation of the Proposed Change

Attachments:
1. Marked-Up Technical Specifications Pages
2. Marked-Up Technical Specifications Bases Pages (For Information Only)

cc: (all with Enclosure/Attachments)

L. Dudes, Regional Administrator USNRC Region II
J. Zeiler, NRC Senior Resident Inspector
T. Hood, NRR Project Manager
L. Haeg, NRR Project Manager

A. Wilson, Attorney General (SC)
R. S. Mack, Assistant Bureau Chief, Bureau of Environmental Health Services (SC)
L. Garner, Manager, Radioactive and Infectious Waste Management Section (SC)

Enclosure

EVALUATION OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

Duke Energy Progress, LLC (Duke Energy) proposes an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (RNP), Unit No. 2. The proposed amendment would add a new function to TS 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation” Table 3.3.2-1 for Feedwater Isolation on Steam Generator (SG) level high-high (i.e., SG overfill protection). In addition, proposed revisions to TS 2.1.1.1 and TS 5.6.5.b are included to reflect the removal of analytical methods no longer applicable for the determination of RNP core operating limits.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

RNP is a Westinghouse-design three loop pressurized water reactor (PWR). The feedwater system is designed to supply water to the SGs under all operating conditions. During normal power operation, the main feedwater (MFW) pumps are utilized to supply the needed water. During periods of shutdown or abnormal conditions, one steam driven and two motor operated auxiliary feedwater pumps may be used.

One main feedwater isolation valve (MFIV), one main feedwater regulation valve (MFRV), and two bypass valves are located on each MFW line, outside but close to containment. The bypass line, with two bypass valves, bypasses both the MFIV and the MFRV. The MFRV and bypass valves in the feedwater line to each SG maintain the proper water level in the SGs for all load conditions. Closure of the MFIVs or MFRVs, and bypass valves, terminates the addition of feedwater to the associated SG. The RNP feedwater system is further described in RNP Updated Final Safety Analysis Report (UFSAR) Section 10.4.6.

When the level in any SG exceeds the high-high water level setpoint of 75% (also referred to at RNP as the high SG level valve interlock), existing SG overfill protection equipment responds by tripping the main turbine, tripping the MFW pumps, and closing the MFRV and bypass valves to the affected SG. The automatic feedwater isolation terminates the overcooling of the primary system and prevents liquid from entering the main steam lines.

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents. UFSAR Section 7.3 describes the Engineered Safety Features (ESF) systems. The existing SG overfill protection function is not part of the ESFAS; however, consistent with the ESFAS, the existing SG overfill protection instrumentation utilizes a 2 out of 3 channel initiating logic that is safety related.

Background

Generic Letter (GL) 89-19, “Request for Action Related to Resolution of Unresolved Safety Issue A-47 “Safety Implication of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f),” provided recommendations concerning automatic SG overfill protection for all PWR plants. The GL recommended that the design of the SG overfill protection system should be sufficiently separate from the MFW control system. The GL also recommended that plant procedures and TS for all plants should include provisions to periodically verify operability of the overfill protection and to assure that automatic overfill protection is available to mitigate MFW overfeed events during reactor power operation. The RNP response to GL 89-19 was submitted to the Nuclear Regulatory Commission (NRC) by letters dated March 19, 1990 (Reference 1) and May 7, 1991 (Reference 2). The Reference 1 response described how the design of RNP's SG overfill protection system met the intent of the GL 89-19 recommendations and also stated: "As recommended by the Generic Letter, HBR2 [H. B. Robinson Steam Electric Plant, Unit No. 2] plant procedures and technical specifications include provisions to periodically verify the operability of the MFW overfill protection and ensure that the automatic overfill protection is operable during reactor power operation (Table 4.1-1, Technical Specifications for HBR2, Item 11). The HBR2 technical specifications include appropriate LCOs [Limiting Conditions for Operation] for this purpose and are commensurate with technical specification requirements for channels that initiate protective action." The Reference 2 response further verified adequate cable separation by walkdown. By letter dated June 24, 1991 (Reference 3), the NRC agreed that RNP's response met the intent of GL 89-19.

The RNP TS were converted to improved standard TS (STS) based on NUREG-1431, "Standard Technical Specifications Westinghouse Plants," as approved by NRC safety evaluation dated October 24, 1997 (Reference 4). The feedwater isolation on SG water level high-high was not included in the new TS 3.3.2. Justification in the STS conversion package stated: "This Function is not classified as an Engineered Safety Feature in the plant design basis and current licensing basis."

RNP UFSAR Section 15.1.2, "Feedwater System Malfunctions that Result in an Increase in Feedwater Flow," currently does not contain an explicit transient analysis for the increase in feedwater flow event. The disposition of this event uses a comparison of a hand calculated increase in heat removal capacity to demonstrate that this transient is bounded by the UFSAR Section 15.1.3 increase in steam flow event when analyzed at power. A comparison of a hand-calculated reactivity insertion rate is used to demonstrate that this transient is bounded by the UFSAR Section 15.4.1 rod withdrawal from subcritical event when analyzed at startup conditions. While these statements remain valid, that evaluation does not acknowledge that the feedwater overfeed must eventually be terminated. The disposition for this event was added to the UFSAR in Cycle 10 following SG replacement with the submittal of Exxon Nuclear Company's stand-alone Chapter 15 analysis methodology (XN-NF-84-74, "Plant Transient Analysis for H. B. Robinson Unit 2 at 2300 MWt with Increased $F_{\Delta H}$"). That methodology also references another Exxon report containing the disposition of Chapter 15 events (XN-NF-83-72, "H. B. Robinson Unit 2, Cycle 10 Safety Analysis Report," Supplement 1). The disposition of the UFSAR Section 15.1.2 event as it appears in the current UFSAR is contained in XN-NF-83-72, Supplement 1 and was approved by the NRC per safety evaluation dated November 7, 1984 (Reference 5).

2.2 <u>Current Technical Specifications Requirements</u>

SG Overfill Protection
TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," specifies that the ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE. TS Table 3.3.2-1 provides a list of all ESFAS functions and specifies the following for each function: APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS, REQUIRED CHANNELS, CONDITIONS, SURVEILLANCE REQUIREMENTS, ALLOWABLE VALUE, and NOMINAL TRIP SETPOINT.

Remove Obsolete Methods
TS 2.1.1.1 specifies the Safety Limit (SL) for the departure from nucleate boiling ratio (DNBR). Specifically. TS 2.1.1.1 currently states:

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

TS 5.6.5, "Core Operating Limits Report (COLR)," contains requirements related to core operating limits established prior to each reload cycle. TS 5.6.5.b provides a listing of analytical methods used to determine the RNP core operating limits.

2.3 Reason for the Proposed Change

SG Overfill Protection

While recently reevaluating the RNP UFSAR Chapter 15.1.2 increase in feedwater flow transient using Duke Energy's in-house methods approved by the NRC in 2018 (Reference 8), it was recognized that SG overfill protection is needed to mitigate this event. A SG overfill event could potentially lead to either a steam line break that challenges the containment fission product barrier should primary-to-secondary leakage be present or fuel failure (i.e., the clad fission product barrier) as a result of a steam line break or stuck open SG relief valve. The concerns with SG overfill as stated in GL 89-19 include the following: "(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."

An increase in feedwater flow can overfill a SG within approximately 5 minutes of the initiation of the event, and therefore automatic action is needed to terminate the event. The high-high SG level trip will prevent SG overfill and ensure that the acceptance criteria for the UFSAR Chapter 15.1.2 event are met. Therefore, the concerns presented by GL 89-19 remain valid and the high-high SG level trip should not have been removed from the TS as part of the STS conversion. This amendment request restores compliance with GL 89-19.

Remove Obsolete Methods

The proposed change would revise the RNP TS to reflect removal of analytical methods that will no longer be used to determine the core operating limits, as these methods were replaced upon transitioning to NRC-approved Duke Energy methods.

2.4 Description of the Proposed Change

SG Overfill Protection

The proposed change adds new feedwater isolation function 5.c "SG Water Level – High-High" to TS Table 3.3.2-1, with the following requirements:

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS: 1,2[(f)],3[(f)]
(Existing note (f): Except when all MFIVs, MFRVs, and bypass valves are closed or isolated by a closed manual valve.)

REQUIRED CHANNELS: 3 per SG

CONDITIONS: D

SURVEILLANCE REQUIREMENTS: SR 3.3.2.1, SR 3.3.2.4, SR 3.3.2.7

ALLOWABLE VALUE: ≤76.16%

NOMINAL TRIP SETPOINT: 75%

There are no physical changes to the plant required as a result of the proposed change. Despite the removal of this function from the TS as part of the STS conversion, the instrumentation has remained functional in the plant and has been regularly tested to ensure that it will actuate if called upon during an event. This proposed change simply formalizes the operability and testing requirements of the SG overfill protection system and prescribes the required actions if the SG overfill protection function is inoperable. The proposed change does not classify the SG overfill protection system as ESFAS equipment; the SG overfill protection function is added to the TS ESFAS table to restore compliance with GL 89-19. The SG overfill protection function is added to the RNP TS without requiring a modification to the current plant design and will be credited in the licensing basis analysis for UFSAR Section 15.1.2. All other transient and accident analyses in the RNP UFSAR are unaffected by this change.

Remove Obsolete Methods

The proposed change to TS 2.1.1.1 removes the XNB departure from nucleate boiling (DNB) correlation. Specifically, TS 2.1.1.1 is proposed to state the following:

> The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP correlation.

The proposed change to TS 5.6.5.b removes certain analytical methods. Specifically, the methods listed below are to be replaced with “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.

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

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

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

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.

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.

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

3.0 TECHNICAL EVALUATION

SG Overfill Protection

Automatic feedwater isolation on high-high SG level is not explicitly credited in any UFSAR Chapter 15 event. The UFSAR Section 15.1.2 increase in feedwater flow event currently contains a disposition that compares the increase in cooling demand to that of the UFSAR 15.1.3 increase in steam flow event when analyzed at power. The disposition of the event initiated from startup conditions compares the reactivity insertion rate to that of the UFSAR 15.4.1 uncontrolled rod withdrawal from subcritical event. While the increase in feedwater flow event is indeed bounded by these other two events, the evaluation did not consider that the feedwater overfeed must eventually be terminated to prevent SG overfill. Without automatic isolation of main feedwater, SG overfill occurs within approximately 5 minutes of the initiation of the event from hot full power. Feedwater overfeeds that are initiated from lower power levels overfill even more quickly. Once the SG overfills, water enters the steam line and introduces the potential for liquid relief through a SG power operated relief valve (PORV) or safety valve. These secondary safety valves may stick open following liquid relief, which could cause this UFSAR Chapter 15 Condition II event to transition to a more serious Condition IV event.

The UFSAR Section 15.1.2 increase in feedwater flow event has been reevaluated using DPC-NE-3009-P-A, “FSAR/UFSAR Chapter 15 Transient Analysis Methodology,” approved by the NRC per safety evaluation dated April 10, 2018 (Reference 8). The transient is initiated by a failed-open MFRV to one SG, increasing MFW flow to that SG. The increase in heat removal by the secondary system causes a decrease in primary system temperature. This event analysis conservatively assumes feedwater isolation at a bounding value of 97% SG level (this analytical limit is discussed further below). This event is most limiting at end-of-cycle due to the more negative moderator temperature coefficient which yields a larger power increase due to the overcooling. Although the Reactor Protection System (RPS) is available to mitigate these effects, a reactor trip signal would not be received prior to reaching the high-high SG level trip setpoint. In addition, although the high-high SG level trip would trip the turbine, this event analysis conservatively does not credit this trip. For the hot full power case, a new steady state power level of approximately 105% rated thermal power (RTP) is reached that is sustained until the high-high SG level setpoint is reached at approximately 2 minutes. For the hot zero power case, a return to power occurs due to negative moderator temperature feedback. The high-high SG level setpoint is reached more quickly than the hot full power case due to the higher initial SG mass and lower primary system heat load. In both cases, feedwater isolation occurs upon reaching the uncertainty-adjusted high-high SG level setpoint and terminates the overcooling event. The results of this analysis show that there is significant margin to the DNB and centerline fuel melt (CFM) acceptance criteria. Peak primary and peak secondary pressures are bounded by the loss of external electrical load transient (UFSAR Section 15.2.2) as detailed in the safety evaluation for DPC-NE-3009 (Reference 8).

The trip setpoint corresponds to the nominal value at which a device is set and expected to change state. The allowable value is the maximum region associated about a setpoint that is still considered to be acceptable for the instrument to fulfill its function without risking exceeding the analytical limit. The function of the high-high SG level setpoint is to isolate MFW before the SG is full of water. Therefore, this is an increasing setpoint and is computed using negative total loop uncertainties. The following equation is used to calculate the maximum value for this setpoint:

$SP_{limit} \leq AL - TLU$

where,

SP_{limit} = calculated setpoint limit
AL = analytical limit
TLU = total loop uncertainty

Modules in the high-high SG level setpoint loop considered in the calculation of TLU consist of a transmitter and comparator. The negative TLU associated with this setpoint is −16.73% span, which considers effects of reference accuracy, calibration tolerance, drift, measurement and test equipment (M&TE) effect, static pressure effect, temperature effect, power supply effect, seismic effect, process measurement effect, and analyzed drift bias. The random component of this loop is ± 2.49% span and the bias component is −14.24% span. Random components are combined using the Square-Root-Sum-of-the-Squares (SRSS) method. The random uncertainty can be reduced by applying the single side of interest and 2sigma Reduction (0.8225 for an original symmetric value based on 2 sigma members as discussed in Section 7.3 and Annex J, Section J.1 of Reference 15). Therefore, the new random loop uncertainty for this setpoint is ± 2.05% span (0.8225 x 2.49% span). This results in a new negative TLU of −16.29% span (−2.05% span − 14.24% span).

In 2002, Westinghouse issued Nuclear Safety Advisory Letter NSAL-02-4 (Reference 10) regarding the void content of the two-phase mixture above the mid-deck plate located between the upper and lower taps used for SG level measurements. The SG narrow range level instrument channels will not indicate water level as accurately as presumed when level is above the mid-deck plate due to these effects. NSAL-02-4 provides a means of determining the “maximum reliable indicated level (MRIL)” to ensure that the high-high SG level trip will protect against the SG becoming water solid and feedwater entering the main steam lines. Using NSAL-02-4, the SG MRIL was determined to be 97.77% narrow range level. Since this is the upper limit for reliable level indication, the analytical limit was conservatively chosen at 97% span. Therefore,

$SP_{limit} \leq 97\%$ span − 16.29% span
$SP_{limit} \leq 80.71\%$ span

Therefore, the proposed high-high SG level setpoint (75%) provides margin to the calculated setpoint limit (80.71%). The proposed “SG Water Level – High-High” nominal setpoint of 75% narrow range level is unchanged from the current plant setpoint. This nominal trip setpoint has been found to be acceptable based on plant operating history and experience. In addition, as described above, the analysis of the UFSAR Section 15.1.2 increase in feedwater flow transient performed with Duke Energy's NRC-approved methods demonstrates that all acceptance criteria are met. The calculations performed in accordance with the Duke Energy setpoint methodology procedure confirmed the current nominal setpoint of 75% as being acceptable.

The as-found calibration tolerance band and as-left calibration tolerance band are calculated and tested consistent with the methodology for all other ESFAS setpoints as specified in the background section of RNP TS Bases 3.3.2 for trip setpoints and allowable values. As discussed in this section of TS Bases, a channel is required to be adjusted if the actual trip setpoint is found outside the as-found calibration tolerance band, such that the actual trip setpoint is within the as-left calibration tolerance band. In addition, the transmitter/sensor is not included in tolerance calculations for this allowable value (AV) because testing consists of a simulated signal injected in place of the field instrument signal. Therefore, the Group As-Found Tolerance (GAFT) used to calculate the AV consists of only the As-Found Tolerance (AFT) of the comparator. This value is 1.16% span, which includes effects of calibration tolerance, drift, and M&TE uncertainties combined using the SRSS method. The As-Left Tolerance (ALT) is 0.50% span, which includes effects of calibration tolerance of the comparator. The AV associated with this setpoint is computed as follows:

AV ≤ SP + GAFT, where SP = calibrated setpoint
AV ≤ 75% Span + 1.16% Span
AV ≤ 76.16% Span

Note that the analytical limit and uncertainty values discussed above (other than the 75% TS nominal trip setpoint and 76.16% TS allowable value) are included to support the justification of this amendment request but remain subject to the provisions of 10 CFR 50.59.

The proposed applicable MODES for the "SG Water Level – High-High" function are MODES 1, 2, and 3. The exception in MODES 2 and 3 is 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. The MFW System may be in operation in Mode 4 during unit startup and shutdown evolutions. One MFW pump is typically started during the heatup as RCS temperature approaches 350°F with flow through the MFW bypass valves and the MFW regulating valves closed. During cooldown, one MFW pump is typically used with flow through the MFW bypass valves and the MFRVs closed until the transition to Residual Heat Removal can be achieved. This transition normally occurs in Mode 3 but may occur in Mode 4. These evolutions are short in duration and happen very infrequently. Therefore, the probability of a feedwater overfeed event occurring in Mode 4 is sufficiently small that an explicit analysis is not performed at these conditions. If a feedwater overfeed occurs in Mode 4 resulting in a steam line break or stuck open safety valve, there is insufficient energy in the RCS or SGs to lead to significant RCS overcooling or containment pressurization. This is consistent with the applicable modes for the MFRVs and bypass valves that are closed on a SG level high-high signal as discussed in the applicability section of the TS Bases for Section 3.7.3. Additionally, the MFW system is not in service in Modes 5 and 6. Therefore, this function is not required to be OPERABLE in Modes 4, 5, and 6. The proposed MODES are consistent with NUREG-1431.

The "SG Water Level – High-High" function utilizes a 2 out of 3 channel initiating logic. The existing Condition D for one channel inoperable (and associated Required Action and Completion Time) is proposed for this function, as it is specific to channel operability.

Furthermore, as stated in the RNP TS Bases, Condition D applies to functions that operate on a 2 out of 3 logic. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, the function is no longer required OPERABLE. The Action for Condition D is modified by a Note that allows a channel for Function 4.c (Steam Line Isolation – Containment Pressure – High High) to be taken out of the trip condition for 6 hours for maintenance purposes. The “Containment Pressure - High High” channels are uniquely designed in that they energize to actuate (perform trip). Maintenance activities that interrupt power to the channel cause the channel to be taken out of the trip condition. Therefore, the note allows conducting these activities without being required to implement extraordinary measures to maintain the channel in the tripped condition. The “SG Water Level – High-High” channels de-energize to actuate (perform trip); therefore, the existing Note in the Action for Condition D does not apply to the “SG Water Level – High-High” function. The corresponding Condition in NUREG-1431 (Condition D) only specifies one Required Action: to place the channel in trip within 72 hours. The proposed Required Actions and Completion Times are more conservative than NUREG-1431.

The proposed SRs for the “SG Water Level – High-High” protection logic are to perform a CHANNEL CHECK (SR 3.3.2.1), CHANNEL OPERATIONAL TEST (SR 3.3.2.4), and CHANNEL CALIBRATION (SR 3.3.2.7), with the frequency controlled by the Surveillance Frequency Control Program (SFCP). Note that the current frequencies of these existing SRs in the SFCP are 12 hours (SR 3.3.2.1), 92 days (SR 3.3.2.4), and 24 months (SR 3.3.2.7). The selection of SRs is consistent with the SRs specified for other existing ESFAS protection functions that require channel operability (e.g., Table 3.3.2-1, items 1d, 1e, 1f, 1g, 4c, 4d, and 4e) as well as the corresponding SRs specified in NUREG-1431 (SR 3.3.2.1, SR 3.3.2.5, and SR 3.3.2.9; note that NUREG-1431 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). The selection of the surveillance interval frequency ensures that provisions are included to periodically verify the operability of the SG overfill protection and ensure overfill protection is operable during startup and reactor power operation. Although this trip function is not included in the TS at the present time, the “SG Water Level – High-High” protection circuitry is currently tested as part of the plant ESFAS surveillance procedure for the other protection functions listed in Table 3.3.2-1. Feedwater isolation testing, which includes tripping of the feedwater pumps and closure of the MFRVs and bypass valves, is performed with the unit in cold shutdown in accordance with the applicable ESFAS integrated test procedures.

All three channels of SG narrow range level indication are used for both SG overfill protection and feedwater control. The feedwater control system uses the three narrow range level signals that are provided via a median selector. Signal isolation devices are present for each channel to ensure adequate separation of the overfill protection and feedwater control system. The power supplies for the feedwater control system and SG overfill protection are fed through different circuits off the same instrument buses. Since power to these instrument buses is provided from a reliable source, with an alternate power supply available during all plant conditions, power supply through different circuits to these two systems are considered separate. Rack components (e.g., the comparator) for the overfill protection and feedwater control systems are located in different racks (i.e. not in the same cabinet). The SG overfill protection cables for different channels do not share common routing, and cable separation is such that two SG overfill protection channels remain operable in the event of a fire in the control portion of the feedwater control system.

Remove Obsolete Methods
RNP TS 5.6.5.b lists the methodologies approved for use in the design and safety analysis of core reloads, with the RNP COLR identifying the methods and revisions used each cycle. The NRC has approved the following Duke Energy methodologies for use by RNP to perform the respective analyses in-house:

- DPC-NE-2005-P, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology," by letter dated March 8, 2016 (Reference 6)
- DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," by letter dated May 18, 2017 (Reference 7)
- DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," by letter dated May 18, 2017 (Reference 7)
- DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," by letter dated May 18, 2017 (Reference 7)
- DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," by letter dated April 10, 2018 (Reference 8)
- DPC-NE-3009-P, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," by letter dated April 10, 2018 (Reference 8)

Additionally, by letter dated April 29, 2019 (Reference 9), the NRC approved the addition of NRC-approved methodology BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," to the list of analytical methodologies in RNP TS 5.6.5.b. The NRC also approved the revision of the fuel centerline melt safety limit to that used in the COPERNIC code as to allow Duke Energy the ability to self-perform fuel rod mechanical analyses for RNP.

The list of COLR methodologies in RNP TS 5.6.5.b has been reviewed to identify those methodologies that have been rendered obsolete by the NRC-approved Duke Energy methodologies listed above, as well as BAW-10231P-A. The obsolete methodologies as well as the XNB correlation DNB limit identified in Section 2.4 of this license amendment request (LAR) are no longer planned for use in the design and safety analysis of core reloads (note that the XNB correlation DNB limit being removed corresponds to the removal of TS 5.6.5.b method number 9, "XNB Critical Heat Flux Correlation"). Analyses completed using the remaining methodologies will continue to provide assurance that the design and safety analysis of the core reloads remain within the TS core operating limits to ensure safe operation. As such, an administrative change is requested to remove this obsolete content.

4.0 <u>REGULATORY EVALUATION</u>

4.1 <u>Applicable Regulatory Requirements/Criteria</u>

10 CFR 50.36(c)(2), "Limiting conditions for operation" states the following, in part:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

10 CFR 50.36(c)(2)(ii) Criterion 3 states the following:

> A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier

10 CFR 50.36(c)(3), "Surveillance requirements" states the following:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

10 CFR 50.36(c)(5), "Administrative controls" states the following:

> Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4.

GL 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f)," provided recommendations concerning automatic SG overfill protection for all PWR plants. It recommended that sufficient separation should be provided between the MFW control system and the SG overfill protection system. It also recommended that plant procedures and TS include provisions to periodically verify the operability of the SG overfill protection system. Upon implementation of this amendment request, RNP will once again meet the intent of the GL 89-19 recommendations, consistent with the original RNP GL response that the NRC found acceptable.

NUREG-1431, "Standard Technical Specifications Westinghouse Plants," is the NRC's approved standard TS for Westinghouse-designed nuclear power plants such as RNP. The proposed TS are consistent with (or more conservative than) NUREG-1431 as described in Section 3.0 above.

The proposed change adds the SG overfill protection function to the RNP TS without requiring a modification to the current plant design. This will ensure operability of the SG overfill protection system to prevent the consequences of overfilling a SG.

This change does not affect plant compliance with the above regulations / guidance (or in the case of GL 89-19, it restores compliance) and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 <u>Precedent</u>

SG Overfill Protection

GL 89-19 required response from licensees of all light water reactors and, therefore, the addition or verification of SG overfill protection in TS would have been completed at that time. Of note are Turkey Point Units 3 and 4, which are three loop Westinghouse-designed PWRs with two out of three logic for SG overfill similar to RNP. Turkey Point received approval to add SG overfill to their TS by safety evaluation dated April 28, 1994 (Reference 11). The Turkey Point TS and TS Bases noted that the SG overfill protection is not part of ESFAS but is added to the TS only in accordance with GL 89-19.

Remove Obsolete Methods
The NRC previously approved changes to Shearon Harris Nuclear Power Plant, Unit 1 (HNP) TS 6.9.1.6.2 and RNP TS 5.6.5.b for the removal of analytical methods no longer planned to be used, as per letters dated April 8, 2021 (HNP, Reference 12), March 30, 2012 (HNP, Reference 13), and December 29, 2011 (RNP, Reference 14). The Reference 12 HNP amendment utilized the same NRC approved Duke Energy methods presented in this LAR as basis for removal of obsolete methods. Note that HNP did not need to remove a DNBR XNB correlation limit from HNP's corresponding TS 2.1.1.a, because that limit did not exist in HNP TS 2.1.1.a. Additional changes in Reference 12 related to RCS flow rate and the large break loss-of-coolant accident methodology are outside the scope of this LAR. The Reference 13 and 14 amendments also revised HNP and RNP TS to permit the use of M5 advanced alloy for fuel rod cladding and fuel assembly structural components in future operating cycles, but that is also outside the scope of this LAR.

4.3 <u>No Significant Hazards Consideration Determination</u>

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) is submitting a request for an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (RNP), Unit No. 2. The proposed amendment would add a new function to TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation" Table 3.3.2-1 for Feedwater Isolation on Steam Generator (SG) level high-high (i.e., SG overfill protection). In addition, proposed revisions to TS 2.1.1.1 and TS 5.6.5.b are included to reflect the removal of analytical methods no longer applicable for the determination of RNP core operating limits.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The proposed change adds new TS requirements for the existing SG overfill protection system to ensure the equipment will operate as designed and to specify actions to take in the case of inoperability. An additional administrative change removes obsolete content from TS 2.1.1.1 and TS 5.6.5.b. The proposed changes do not alter the design, configuration, operation, or function of any plant structure, system, or component. The SG overfill protection system prevents SG overfill and ensures that the acceptance criteria for the UFSAR Chapter 15.1.2 event are met. As a result, the outcomes of previously evaluated accidents are unaffected. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

   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 change adds new TS requirements for the existing SG overfill protection system to ensure the equipment will operate as designed and to specify actions to take in the case of inoperability. An additional administrative change removes obsolete content from TS 2.1.1.1 and TS 5.6.5.b. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not challenge the performance or integrity of any safety related system. The proposed changes neither install nor remove any plant equipment, nor alter the design, physical configuration, or mode of operation of any plant structure, system, or component. No physical changes are being made to the plant, so no new accident causal mechanisms are being introduced.

Therefore, the proposed changes do 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.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system boundary, and the containment system. The proposed change adds new TS requirements for the existing SG overfill protection system to ensure the equipment will operate as designed and to specify actions to take in the case of inoperability. An additional administrative change removes obsolete content from TS 2.1.1.1 and TS 5.6.5.b. The proposed changes will have no effect on the availability, operability, or performance of the safety related systems and components. The proposed changes do not alter the design, configuration, operation, or function of any plant structure, system, or component. The ability of any operable structure, system, or component to perform its designated safety function is unaffected by the proposed changes. There is no impact on the fission product barriers or parameters associated with licensed safety limits.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (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 to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or a 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 changes meet 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. Carolina Power & Light Company (CP&L) letter, *Response to NRC Generic Letter 89-19 Request for Action Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants"*, dated March 19, 1990 (ADAMS Accession No. ML14188B683)
2. CP&L letter, *Additional Information Regarding Generic Letter 89-19*, dated May 7, 1991 (ADAMS Accession No. ML14184A829)
3. NRC letter, *Closeout of Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47 'Safety Implication of Control System in LWR Nuclear Power Plants' Pursuant to 10 CFR 50.54(f)" – H. B. Robinson Steam Electric Plant, Unit No. 2 (TAC No. 74990)*, dated June 24, 1991 (ADAMS Accession No. ML14184A839)
4. NRC letter, *Issuance of Amendment No. 176 to Facility Operating License No. DPR-23 Regarding Conversion to Improved Standard Technical Specifications – H. B. Robinson Steam Electric Plant, Unit No. 2 (TAC No. M96440)*, dated October 24, 1997 (ADAMS Accession Nos. ML020560172, ML14175A922, and ML14175A924)
5. NRC letter containing Amendment No. 87 to Facility Operating License No. DPR-23, dated November 7, 1984 (ADAMS Accession No. ML020520268)
6. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Report DPC-NE-2005-P, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology" (CAC Nos. MF5872 and MF5873)*, dated March 8, 2016 (ADAMS Accession Number ML16049A630)
7. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-1008-P Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NF-2010 Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" (CAC Nos. MF6648/MF6649 and MF7693/MF7694)*, dated May 18, 2017 (ADAMS Accession Number ML17102A923)
8. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," and DPC-NE-3009-P, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology" (CAC Nos. MF8439 and MF8440; EPID L-2016-LLA-0012)*, dated April 10, 2018 (ADAMS Accession Number ML18060A401)
9. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications to Support Self-Performance of Core Reload Design and Safety Analyses (EPID L-2017-LLA-0356)*, dated April 29, 2019 (ADAMS Accession Number ML18288A139)

10. Westinghouse Nuclear Safety Advisory Letter NSAL-02-4, *Maximum Reliable Indicated Steam Generator Water Level*, dated February 19, 2002
11. NRC letter, *Turkey Point Units 3 and 4 – Issuance of Amendments Re: Steam Generator Overfill Protection (TAC Nos. M88560 and M88561)*, dated April 28, 1994 (ADAMS Accession No. ML013380368)
12. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment No. 185 Regarding Reduction of Reactor Coolant System Minimum Flow Rate and Update to the Core Operating Limits Report References (EPID L-2020-LLA-0040)*, dated April 8, 2021 (ADAMS Accession No. ML21047A470)
13. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: The Use of AREVA's M5™ Advanced Alloy in Fuel Cladding and Fuel Assembly Components (TAC No. ME5409)*, dated March 30, 2012 (ADAMS Accession No. ML12058A133)
14. NRC letter, *H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Technical Specifications Related to Use of AREVA's M5 Advanced Alloy in Fuel Cladding and Fuel Assembly Components (TAC No. ME4911)*, dated December 29, 2011 (ADAMS Accession Number ML11342A165)
15. International Society of Automation (ISA) RP67.04.02, *Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation*, publication date December 10, 2010

Attachment 1

Marked-Up Technical Specifications Pages

(6 Pages Follow)

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:

Delete text

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.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

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.					
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%
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.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. Axial Flux Difference (AFD) limits for Specification 3.2.3;

8. Boron Concentration limit for Specification 3.9.1;

9. Reactor Core Safety Limits Figure for Specification 2.1.1;

10. Overtemperature ΔT and Overpower ΔT setpoint parameter values for Specification 3.3.1; and

11. Reactor Coolant System pressure, temperature and flow Departure from Nucleate Boiling (DNB) limits for Specification 3.4.1.

12. ECCS Accumulators boron concentration limits for Specification 3.5.1.

13. ECCS Refueling Water Storage Tank boron concentration limits for Specification 3.5.4.

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.~~

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

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)

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements (continued)

Replace text in items 9 and 11 with "Deleted"

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

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

(continued)

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.~~

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

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.

(continued)

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.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(continued)

Attachment 2

Marked-Up Technical Specifications Bases Pages (For Information Only)

(3 Pages Follow)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

d, e. Steam Line Isolation - High Steam Flow in Two Steam Lines Coincident with T_{ave} - Low or Coincident With Steam Line Pressure - Low

These Functions (4.d and 4.e) provide closure of the MSIVs during an SLB or inadvertent opening of an SG relief or a safety valve, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.

These Functions were discussed previously as Functions 1.f. and 1.g.

These Functions must be OPERABLE in MODES 1 and 2, and in MODE 3, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines unless all MSIVs are closed. These Functions are not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

5. Feedwater Isolation

carryover of water into the steam lines and

The primary function of the Feedwater Isolation signal is to stop the excessive flow of feedwater into the SGs. This Function is necessary to mitigate the effects of overfeeding the SGs, which could result in excessive cooldown of the primary system.

The Function is actuated on an SI signal and performs the following functions:

a high-high SG level or

- Trips the MFW pumps; and

(SI signal only)

- Shuts the MFW isolation valves, MFW regulating valves and the bypass feedwater regulating valves.

The high-high SG level signal also trips the main turbine.

~~This Function is actuated by an SI signal.~~ The RPS initiates a turbine trip signal whenever a reactor trip is generated. In the event of SI, the unit and the turbine generator are tripped by the RPS. The MFW System is also taken out of operation and the AFW

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

5. Feedwater Isolation (continued)

System is automatically started. The SI signal was discussed previously.

a. Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Feedwater Isolation - Safety Injection

Feedwater Isolation is also initiated by all Functions that initiate SI. T he Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI -function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

c. Feedwater Isolation - SG Water Level - High-High

This signal provides protection against excessive feedwater flow. The feedwater control system uses the three narrow range level signals that are provided via a median selector; only three protection channels are necessary to satisfy the protective requirements. Each SG is considered a separate Function for the purpose of this LCO. The setpoints are based on percent of narrow range instrument span.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Nominal Trip Setpoint reflects only steady state instrument uncertainties.

This function is not part of the ESFAS and is added to the Technical Specifications to protect against SG overfill as discussed in NRC Generic Letter 89-19.

Feedwater Isolation Functions must be OPERABLE in MODES 1, 2, 3 and 4 (Mode 4 is SI Only) except when all MFIVs, MFRVs, and associated bypass valves are closed or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES 5 and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

6. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

from full power conditions in an orderly manner and without challenging unit systems.

D.1, D.2.1, and D.2.2

Condition D applies to:

- Pressurizer Pressure - Low;
- Steam Line Differential Pressure - High;
- High Steam Flow in Two Steam Lines Coincident With T_{avg} - Low or Coincident With Steam Line Pressure - Low; ~~and~~
- Steam Line Isolation Containment Pressure - High High~~.~~ ; and
- SG Water Level - High-High

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Action for Condition D is modified by a Note that allows a channel for Function 4.c, Steam Line Isolation – Containment Pressure – High High , to be taken out of the trip condition for 6 hours for maintenance purposes. The channel may be taken out of the trip condition multiple times provided the total time out of trip does not exceed 6 hours (not including the initial 6 hour action time). The Containment Pressure - High High channels are uniquely designed in that they are required to be

(continued)