ATTACHMENT 1

Marked-up Technical Specification Pages:

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EJW1/018/3

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Variable Power Level - High(1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of < 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure - High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure - High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure - Low	≥ 626.0 psia (2)	≥ 621.0 psia (2)
7. Steam Generator Pressure Difference - High (Logic in TH/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level - Low	≥ 39.5% (3)	≥ 39.1% (3)

20.5 %

19.5 %

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
6. LOSS OF POWER		
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3120 volts	≥ 3120 volts
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	≥ 360 volts	≥ 360 volts
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3848 volts with a 10-second time delay	≥ 3848 volts with a 10-second time delay
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	≥ 432 volts	≥ 432 volts
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Steam Generator ΔP-High	≤ 180.0 psid	≤ 187.5 psid
d. SG 2A&2B Level Low	≥ 20.6% (19.0%)	≥ 20.0% (18.0%)
e. Feedwater Header High ΔP	≤ 100.0 psid	≤ 107.5 psid

BASES

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to or concurrently with a safety injection (SIAS). This also provides assurance that a reactor trip is initiated prior to or concurrently with an MSIS.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint of 620 psia is sufficiently below the full load operating point of approximately 885 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of 30 psi in the safety analyses.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required. ADD

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower excore neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15% power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

THIS TRIP ALSO PROTECTS AGAINST VIOLATION OF THE SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS (SAFDL) FOR DNBR, OFFSITE DOSE AND THE LOSS OF SHUTDOWN MARGIN FOR ASYMMETRIC STEAM GENERATOR TRANSIENTS SUCH AS THE OPENING OF A MAIN STEAM SAFETY VALVE OR ATMOSPHERIC DUMP VALVE.

ATTACHMENT 2

SAFETY EVALUATION

Introduction

This is a request to revise Technical Specification 2.2.1, Reactor Protective Instrumentation (RPS), and 3/4.3.2, Engineered Safety Features Actuation System (ESFAS) Instrumentation and their associated Bases of the Technical Specifications for St. Lucie Unit 2.

Discussion

The existing Technical Specification setpoints correspond to a Cycle 1 analysis assumption of steam generator low level RPS setpoint of 30% narrow range (NR) for the most limiting event, Loss of Feedwater, and 5% NR setpoint for all accidents. The difference between the analysis assumption and specified setpoint (9.5%) accounts for process equipment uncertainties, equipment response times and RPS cabinet uncertainties. Auxiliary feedwater was manually actuated by the Reactor Operator, as reflected in all the analyses presented for Cycle 1. The safety grade automatic auxiliary feedwater actuation system (AFAS) was installed before Cycle 1 operation to satisfy a post-TMI requirement, NUREG-0737, Item II.E.1.2.

This proposed change does not impact the results of the Safety Analysis presented for Cycles 2 or 3 but only reflects a change in the analysis assumptions made for Cycle 1 versus Cycles 2 or 3. The new values take into consideration the assumed RPS and ESFAS steam generator low level analysis setpoint of $\leq$ 5% NR and the related instrument uncertainties. The calculated uncertainties were derived using accepted methodology for instrument uncertainties calculations (CEN-112(s) Rev. 0, "Plant Protection System - Selection of Trip Setpoint Values," November 15, 1979), which takes into account inherent process instrumentation errors, equipment response time, instrument drift and environmental concerns for accident conditions, as appropriate. The proposed setpoint numbers are applicable for operation of the RPS and ESFAS systems and are bounding for either the existing Barton M764 transmitters or Rosemount 1154 transmitters (possible replacement transmitters).

The proposed change contains those Technical Specifications needed to support the reduction of the low steam generator level setpoint for actuation of reactor trip and initiation of the AFAS in the RPS and ESFAS.

The proposed change revises Table 2.2-1 Reactor Protective Instrumentation Trip Setpoint Limits and its associated Bases, to reflect a reduction in the setpoint and allowable values for the reactor trip on steam generator level - low.

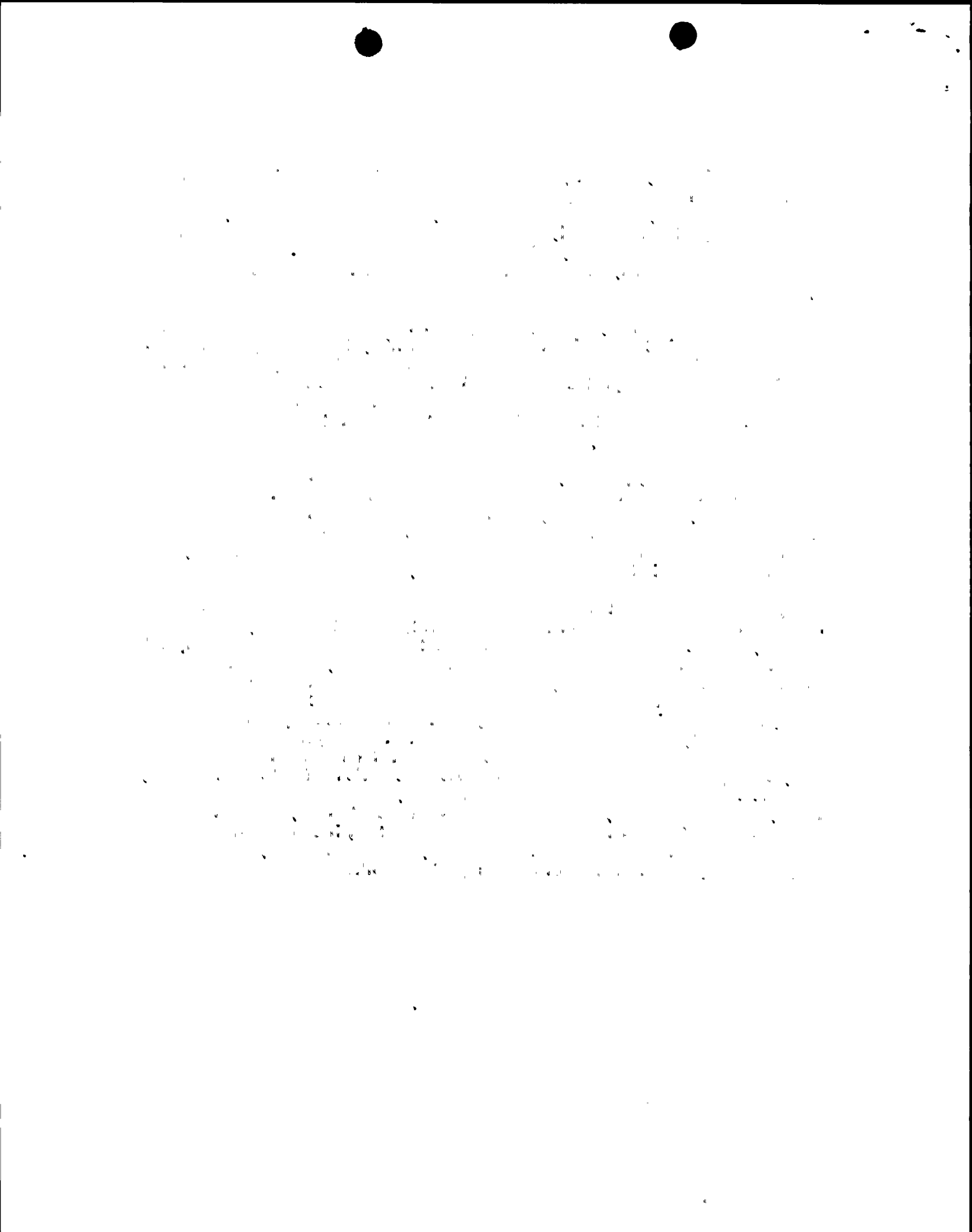
The proposed change also revises Table 3.3-4 Engineered Safety Features Actuation System Instrumentation Trip Values, to reflect a reduction in the initiation setpoint and allowable value for the AFAS on steam generator level - low.

Table 2.2-1, item 8, presently specifies that the trip setpoint on low steam generator level is $\geq$ 39.5% NR, with an allowable value of $\geq$ 39.1%. These values are based on the Cycle 1 Final Safety Analysis Report (FSAR) analysis. The proposed change to Table 2.2.-1, item 8 decreases the trip setpoint on low steam generator level to $\geq$ 20.5% NR, with an allowable value of $\geq$ 19.5%. These values correspond with an analytical setpoint of 5% used in the accident analysis for Cycles 2 and 3.

The Bases for Table 2.2.1, item 8, specify that the trip on low steam generator level protects against exceeding the reactor coolant system (RCS) pressure limit of 110% of design pressure for a Loss of Feedwater event. The proposed change adds that this trip also protects against violation of a specified acceptable fuel design limit (SAFDL) and offsite dose limits for the anticipated operational occurrence of the inadvertent opening of a steam generator atmospheric dump valve or a main steam safety valve.

Table 3.3-4, item 7.d, presently specifies that the AFAS setpoint on low steam generator level is $\geq$ 20.6% NR with an allowable value of $\geq$ 20.0%. These are based on the Cycle 1 FSAR analysis. The proposed change to Table 3.3-4, item 7.d, revises the AFAS setpoint on low steam generator level to $\geq$ 19.0% NR, with an allowable value of $\geq$ 18.0%. These values correspond with an analytical setpoint of 5% used in the accident analysis for Cycles 2 and 3.

The proposed amendment to the low steam generator level setpoints does not impact the results presented in the Reload Safety Evaluation for Cycles 2 and 3 but only reflects changes to the determination of the instrument setpoint consistent with the Cycles 2 and 3 RPS and ESFAS analysis assumptions. The events which can be impacted by changes in RPS and ESFAS steam generator low level setpoints are the Loss of Feedwater event, the Feedwater Line Break (FLB) event, the Steam Line Break event (SLB), and the Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve. The Cycle 2 Safety Analysis assumed a $\leq$ 5% NR RPS and ESFAS steam generator low level setpoint. The analysis setpoint was set at 5% NR to ensure actuation prior to reaching 0%, under accident conditions. The results of the analyses for the events presented in the Cycle 2 Reload Safety Evaluation demonstrated that all key parameters were below the acceptance criteria. Based on this evaluation, it can be concluded that the proposed amendment has been addressed by the existing Analyses of Record and, therefore, there is no impact on the reported results.



ATTACHMENT 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated or (3) involve a signficant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change lowers the probability of previously evaluated events by decreasing the likelihood of an unplanned reactor trip or Auxiliary Feedwater Actuation System (AFAS) initiation on low steam generator level.

(2) Use of the modified specification would not create the possibilityof a new or different kind of accident from any accident previously evaluated.

The setpoints for the Reactor Protective Instrumentation (RPS) trip and the AFAS have been established such that they ensure actuation of these functions before the instrumentation goes off scale. This starts with a minimum level of 5% NR and includes appropriate errors for an inside containment accident such as a Steam Line Break. The difference between the setpoint and allowable values accounts for the instrumentation drift over the specified surveillance interval.

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

As discussed in the Safety Evaluation, the results of all analyses remain within the acceptance criteria of the Standard Review Plan.

Based on the above, we have determined that the amendment request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.