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A4.05
PR

June 22, 1994

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Technical Specification Change Request NPF-38-152

Gentlemen:

The attached description and safety analysis supports a change to the Waterford 3 Technical Specifications (TS). The proposed change will modify the TS by adjusting 3 Plant Protection System (PPS) trip setpoints and allowable values. The change adjusts the affected TS values in a more conservative direction such that they will be consistent with the current setpoint/uncertainty methodology being implemented at Waterford 3.

The new setpoints and allowable values are based on a revised PPS Setpoint Analysis. The new analysis will become effective as Waterford 3 restarts following the current refueling outage. Therefore, Waterford 3 will adhere to the revised conservative values during startup and operation following Refuel 6.

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Should you have any questions or comments concerning this request, please contact Paul Caropino at (504)739-6692.

Very truly yours,

RPBarkhurst

R.P. Barkhurst
Vice President, Operations
Waterford 3

RPB/PLC/ssf

Attachment: Affidavit
NPF-38-152
Reference List
Figure 1

cc: L.J. Callan, NRC Region IV
D.L. Wigginton, NRC-NRR
R.B. McGehee
N.S. Reynolds
NRC Resident Inspectors Office
Administrator Radiation Protection Division
(State of Louisiana)
American Nuclear Insurers

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the matter of | ) | |
| | ) | |
| Entergy Operations, Incorporated | ) | Docket No. 50-382 |
| Waterford 3 Steam Electric Station | ) | |

AFFIDAVIT

R.P. Barkhurst, being duly sworn, hereby deposes and says that he is Vice President Operations - Waterford 3 of Entergy Operations, Incorporated; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached Technical Specification Change Request NPF-38-152; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

R.P. Barkhurst

R.P. Barkhurst
Vice President Operations - Waterford 3

| | | |
|---|---|---|
| STATE OF LOUISIANA | ) | |
| | ) | ss |
| PARISH OF ST. CHARLES | ) | |

Subscribed and sworn to before me, a Notary Public in and for the Parish and State above named this 22nd day of June, 1994.

Steven E. Farr

Notary Public

My Commission expires WITH LIFE.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-38-152

The proposed change affects Technical Specification (TS) Table 2.2-1 REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS, Table 3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES and associated Bases by modifying three parameter setpoints as follows:

1. Table 2.2-1 Item 2) Linear Power Level-High: the TRIP SETPOINT of ≤ 110% of RATED THERMAL POWER and the ALLOWABLE VALUE of ≤ 110.7% RATED THERMAL POWER are changed to reflect ≤ 108% and ≤ 108.76% respectively.

2. Table 2.2-1 Item 4) Pressurizer Pressure-High: the TRIP SETPOINT of ≤ 2365 psia and ALLOWABLE VALUE of ≤ 2372 psia are changed to reflect ≤ 2350 psia and ≤ 2359 psia respectively.

3. Table 3.3-4 Item 7 c.& d.)Emergency Feedwater, Steam Generator ΔP-High: the TRIP VALUE of ≤ 127.6 psid and ALLOWABLE VALUE of ≤ 136.6 psid are changed to ≤ 123 psid and ≤ 134 psid respectively.

The proposed changes are necessary to accommodate additional uncertainty and will have no effect on the original safety analysis setpoints (analytical limits).

Existing Specification

See Attachment A

Proposed Specification

See Attachment B

Background

The proposed change updates the TS consistent with the Waterford 3 revised PPS Setpoint Uncertainty Calculation (Reference 1). The revised calculation was generated using a new Waterford 3 Setpoint and Uncertainty Determination guideline (Reference 2) that provides plant personnel with a description of the governing codes and standards, plant specific criteria, and concepts involved in instrument loop uncertainty analysis and setpoint determination. The Setpoint and Uncertainty Determination guideline was generated to identify and define additional uncertainty factors that have evolved in the industry since the original Waterford 3 PPS Setpoint Analysis (Reference 3) was developed.

The use of improved guidelines is intended to enhance plant safety and in no way invalidates the previous analysis which was also performed using approved methodology.

This proposed change is also driven by a change in plant components. Several transmitters that provide input to the PPS have been upgraded. This upgrade was completed during our recent refueling outage and the new PPS Setpoint Calculation incorporates the new transmitter performance data. An engineering calculation (Reference 4) has demonstrated that the specifications for the original transmitters bound those for the upgraded transmitters.

Description

The proposed change modifies the identified parameter TRIP SETPOINTS and ALLOWABLE VALUES to reflect the following:

| PARAMETER | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| HI Linear Power | ≤ 108 % PWR | ≤ 108.76 % PWR |
| HI Pressurizer Pressure | ≤ 2350 PSIA | ≤ 2359 PSIA |
| HI Steam Generator $\Delta P$ | ≤ 123 PSID | ≤ 134 PSID |

The Reactor Protection System (RPS) Instrumentation Trip Setpoint Limits and the Engineered Safety Feature Actuation System (ESFAS) Instrumentation Trip Values listed in the Technical Specification Tables 2.2-1 and 3.3-4 prescribe those settings for critical parameters that will avoid exceeding any analytical limit for postulated Design Bases Accidents (DBAs) or Anticipated Operational Occurrences (AOOs).

The Trip Setpoints and Allowable Values are based on analytical limits stated in the Final Safety Analysis Report (FSAR). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrumentation drift, and severe environment effects for those RPS and ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49, Allowable Values specified in Tables 2.2-1 and 3.3-4 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the proposed trip setpoints, including their explicit uncertainties, is documented in Reference 1 and 2.

The specific safety analysis applicable to each protective function affected by the proposed change are identified below:

Linear Power Level-High This function provides a reactor trip on high linear neutron flux level. This trip provides protection against core damage during the following:

- Uncontrolled CEA Withdrawal from Low Power (AOO);
- Uncontrolled CEA Withdrawal at Power (AOO); and
- CEA Ejection (Accident).

Pressurizer Pressure-High This function provides a reactor trip on high pressurizer pressure. This trip provides protection for the high Reactor Coolant System pressure Safety Limit. In conjunction with the pressurizer safety valves and the main steam safety valves, it provides protection against overpressurization of the Reactor Coolant Pressure Boundary during the following events:

- Loss of Electrical Load Without a Reactor Trip Being Generated by the Turbine Trip (AOO);
- Loss of Condenser Vacuum (AOO);
- CEA Withdrawal From Low Power Conditions (AOO);
- Chemical and Volume Control System Malfunction (AOO); and
- Main Feedwater System Pipe Break (Accident).

Steam Generator Differential Pressure-High This function of the Plant Protection System enables emergency feedwater flow only to the intact Steam Generator (SG) when the high SG differential pressure condition exist. ESFAS logic includes SG Differential Pressure-High (SG-1 > SG-2 or SG-2 > SG-1) to determine if a rupture in either generator has occurred. Rupture is assumed if the affected generator has a low pressure condition, unless that generator is significantly higher in pressure than the other generator. This latter feature allows feeding the intact SG, even if both are below the Main Steam Isolation Signal (MSIS) setpoint, while preventing the ruptured generator from being fed. Not feeding a ruptured generator prevents containment overpressurization during analyzed events. The most limiting accident condition for the High Steam Generator Differential Pressure function is a Steam Line Break event.

The methodology used in the revised PPS Setpoint Calculation is generally unchanged. However, several factors are now considered or clarified in the revised analysis (e.g. cable insulation resistance degradation, steam generator elongation, containment conditions during specific accidents, and more precise definition of process measurement errors). Figure 1 is provided to illustrate the proposed setpoint change.

## SAFETY ANALYSIS

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   Implementing the proposed change will not affect any design basis accident. The revised trip and actuation setpoints are based upon the same analytical limits that form the basis for the current trip and actuation setpoints. The design basis for each trip and actuation setpoint was verified to be consistent with the appropriate accident analyses as part of the process of revising the PPS setpoint analysis. The proposed changes in trip and actuation setpoints are all in the conservative (away from the analytical limits) direction. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any previously analyzed accident.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously analyzed?

   Response: No

   Plant operation and the manner in which the plant is operated will not be altered as a result of implementing the proposed change since no new system or design change is being implemented. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in margin of safety?

   Response: No

   The current safety margins of the affected trip setpoints and allowable values is preserved by the proposed change. This is assured by retaining the current analytical limit for the affected parameters. Since the analytical limits are not affected and the total channel uncertainty is increased the margin of safety for the affected trip setpoints and allowable values is preserved. Therefore, the proposed change will not involve a significant reduction in margin of safety.

Safety and Significant Hazards Determination

Based on the above safety analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC final environmental statement.

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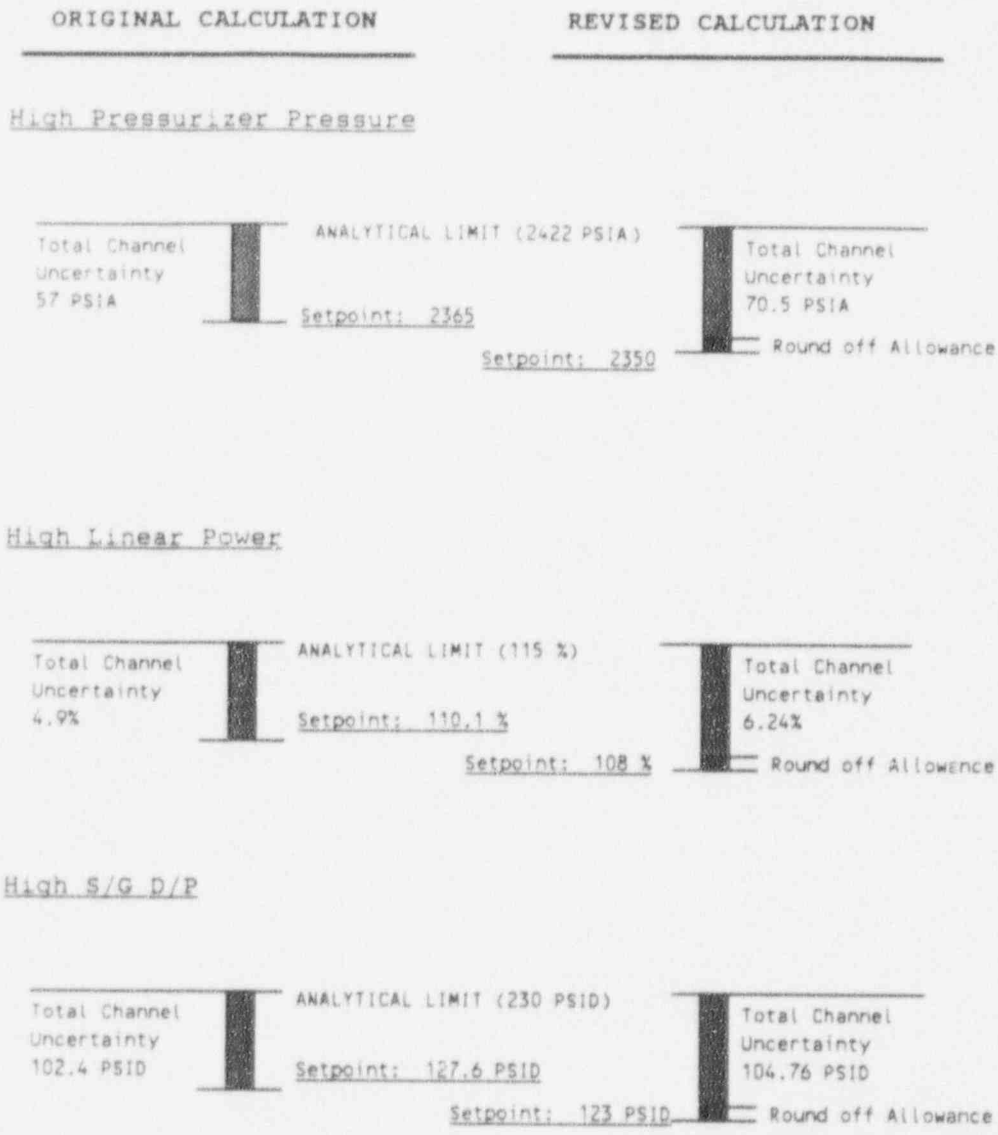
Figure 1



# References


1. Plant Protection System Setpoint Uncertainty Calculation (EC-I92-019 Rev.A) dated August 30,1993.

2. I&C Engineering Design Guide for Setpoint and Uncertainty Determination Rev. 1 dated February 26,1993

3. Combustion Engineering "LP&L WSES-3 PPS Setpoint Analysis" (9270-ICE-36182 Rev.2) dated August 16,1983.

4. Rosemount Transmitter Comparison Calculation (EC-I92-030 Rev.1) dated 06/11/93

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Attachment A