ATTACHMENT 1

St. Lucie Unit 1 Marked-up Technical Specification pages:

B 2-8
2-5
3/4 3-2
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Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspand to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

REPLACE

Loss of Load (Turbine)

The Loss of Load (Turbine) trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip is an equipment protective trip only and is not required for plant safety. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.
9a. Steam Generator Pressure Difference High (1) (logic in TM/LP)	≤ 135 psid	≤ 135 psid
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low ~~(3)~~ DELETE (5) ADD	≥ 800 psig	≥ 800 psig
11. Rate of Change of Power - High (4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

(5) Trip may be bypassed below 25% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 25% of RATED THERMAL POWER.

ADD

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Power Level - High	4	2(a)	3(f)	1, 2	2#
3. Reactor Coolant Flow - Low	4/SG	2(a)/SG	3/SG	1, 2 (e)	2#
4. Pressurizer Pressure - High	4	2	3	1, 2	2#
5. Containment Pressure - High	4	2	3	1, 2	2#
6. Steam Generator Pressure - Low	4/SG	2(b)/SG	3/SG	1, 2	2#
7. Steam Generator Water Level - Low	4/SG	2/SG	3/SG	1, 2	2#
8. Local Power Density - High	4	2(c)	3	1	2#
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2#
9a. Steam Generator Pressure Difference - High	4	2(a)	3	1, 2 (e)	2#
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	4	2(c) DELETE (g) ADD	3	1	2#

[illegible]

CIA → (c)

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL power is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

(e) Deleted

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ADD

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

(g) Trip may be bypassed below 25% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 25% of RATED THERMAL POWER.

ATTACHMENT 2

St. Lucie Unit 2 Marked-up Technical Specification pages:

2-5
2-6
3/4 3-2
3/4 3-3

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High(5)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps-Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power - High(4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow - Low	≥ 95.4% of design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
ADD (6)		
15. Loss of Load (Turbine) Hydraulic Fluid Pressure - Low (8) DELETE	≥ 800 psig	≥ 800 psig

*Design reactor coolant flow with four pumps operating is 363,000 gpm.

**10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when the THERMAL POWER is greater than or equal to 0.5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(3) % of the narrow range steam generator level indication.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

(5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 15% of RATED THERMAL POWER.

(6) Trip may be bypassed below 25% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 25% of RATED THERMAL POWER.

ADD

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	4	2	4	1, 2	1
	4	2	4	3*, 4*, 5*	5
2. Variable Power Level - High	4	2(a)(d)	3	1, 2	2#
3. Pressurizer Pressure - High	4	2	3	1, 2	2#
4. Thermal Margin/Low Pressure	4	2(d)	3	1, 2	2#
5. Containment Pressure - High	4	2	3	1, 2	2#
6. Steam Generator Pressure - Low	4/SG	2/SG(b)	3/SG	1, 2	2#
7. Steam Generator Pressure Difference - High	4	2(a)(d)	3	1, 2	2#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#
9. Local Power Density - High	4	2(c)(d)	3	1	2#
10. Loss of Component Cooling Water to Reactor Coolant Pumps	4	2	3	1, 2	2#
11. Reactor Protection System Logic	4	2	3	1, 2	2#
				3*, 4*, 5*	5
12. Reactor Trip Breakers	4	2(f)	4	1, 2	4
				3*, 4*, 5*	5
13. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating - Rate of Change of Power - High	4	2(e)(g)	3	1, 2	2#
b. Shutdown	4	0	2	3, 4, 5	3
14. Reactor Coolant Flow - Low	4/SG	2/SG(d)	3/SG	1, 2	2#
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	4	2(~~g~~) DELETE ADD (h)	3	1	2#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER in conjunction with (d) below; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 0.5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 15% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL power is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ADD

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

(h) Trip may be bypassed below 25% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 25% of RATED THERMAL POWER.

[illegible]

[illegible]

[illegible]

ATTACHMENT 3

SAFETY ANALYSIS*

1. INTRODUCTION

At the St. Lucie Units 1 and 2, the reactor trip on turbine trip signal is presently enabled at 15% power although the Steam Bypass Control System (SBCS), with its dump capacity of up to 42% of nominal (2700 MW) steam flow, is available to accommodate power levels above this setpoint without requiring a reactor trip. A modification to raise the setpoint at which reactor trip on turbine trip is automatically enabled, has the potential for preventing unnecessary reactor trips caused by turbine trip signals actuated during power ascension maneuvers. A specific plant evolution where this change would be beneficial is during the swap from the 15% capacity bypass valves to the main feedwater regulating valves.

In order to raise the power level at which the reactor trip is enabled on a turbine trip signal, the consequences of a turbine trip without reactor trip event, at the selected new power level, need to be evaluated. Since this is an event that will result in a heatup of the RCS, the evaluation is performed with respect to the following three safety criteria:

- PORV Opening, i.e., potential increase in Small Break Loss of Coolant Accident (SBLOCA) probability (NUREG-0611).
- Primary and Secondary Integrity (System Overpressure)
- Radiological Releases

It has been demonstrated that for normal plant operation with all the control systems assumed operational, a turbine trip without reactor trip event occurring at 25% power level does not result in challenges to the PORVs, system integrity or the environment. This same conclusion also applies when conservative assumptions are made with respect to steam generator tube plugging levels and the main control systems that mitigate the effects of the heatup produced in this event: the Pressurizer Pressure Control System (PPCS) and the Steam Bypass Control System (SBCS).

An evaluation has also been performed, allowing the reactor to eventually trip on low steam generator level following a turbine trip at 25% power. This has been analyzed to verify that there are no adverse consequences even if the operator does not take preventive action to increase feedwater flow to match steam flow.

* This Safety Evaluation applies to Units 1 & 2. Unit 1 differences are shown in [brackets]

The RETRAN02-MOD04 code (Reference 1) has been utilized to perform the analyses in support of this safety evaluation. The FPL RETRAN base models for both St. Lucie Units which include a detailed or multinode steam generator representation have been used in the analyses. Competence with these RETRAN models has been demonstrated in the two FPL RETRAN Topical Reports previously submitted to the NRC (References 2 and 3).

2. SAFETY ANALYSIS BASIS

The turbine trip without reactor trip at 25% power has been analyzed considering the effect of single failures in the control systems that have the most influence in the mitigation of the consequences of this transient. At both St. Lucie Units, the two such control systems are: the Pressurizer Pressure Control System (PPCS) and the Steam Bypass Control System (SBCS). Listed below are the two single failure scenarios for these two systems:

Failure of the pressurizer pressure controller that regulates the pressurizer spray valves resulting in total failure of the spray system to open.

Failure of the Proportional-Integral-Differentiator (PID) controller that regulates the operation of the five valves in the Steam Bypass Control System, during its pressure controlling mode of operation, resulting in the failure of the valves to open.

Of the two above postulated single failures, the worst, in terms of primary pressurization, is that of the PID controller which results in the total unavailability of the SBCS. The consequences of such failure are no greater at 25% power than they are at 15% power. In both cases the RCS heatup results in PORV opening and reactor trip on high pressurizer pressure. Therefore, the potential for a SBLOCA, during a turbine trip without reactor trip event, is not affected by the proposed change of the reactor trip on turbine trip enable from 15% to 25% power level.

Since the single failure of the SBCS does not increase the potential for a SBLOCA potential, the safety analysis has to concentrate on the consequences of the other postulated single failure: that of the pressurizer spray system, during a turbine trip without reactor trip at 25% power.

The potential overpressure consequences of this transient have been evaluated to verify that for this type of scenario (Decreased Heat Removal by the Secondary Side) the current limiting Category II Event remains the Loss of Condenser Vacuum [Loss of External Load] evaluated in the respective FSARs.

The assessment of the radiological consequences of the transient is not applicable because the scenario analyzed here does not involve radiological releases to the atmosphere.

3. INITIAL CONDITIONS AND ASSUMPTIONS OF BASE ANALYSIS

The base analysis to support the change of the reactor trip on turbine trip enable from 15% to 25% has been initialized at the conditions shown in Table 1. A justification for the most relevant parameters/setpoints or initial conditions in the table is provided below.

Power Level

Although the proposed change in the reactor trip on turbine trip enable setpoint is from 15% to 25%, the initial power level has been assumed at 27% to include the typical safety analysis 2% calorimetric power uncertainty.

Reactor Trip

The proposed change in this safety analysis extends the power level at which reactor trip is enabled, upon turbine trip, from 15% to 25% power level. In order to maximize the RCS pressurization in the analysis, this trip has been ignored. The results of the analysis performed in this way envelop all possible scenarios at different power levels and reactivity coefficients in the 15% to 25% range. That is, if the PORVs are not challenged in the analyzed Turbine Trip without Reactor Trip scenario at 25% power, it can be concluded that they will not be challenged at lower power levels during the same type of scenario.

Primary System Flow

The minimum value of 363,000 gpm [370,000] allowed by the respective Technical Specifications has been assumed.

Reactivity Parameters

UNIT 2
The physics parameters used in the analysis are based on Combustion Engineering (CE) supplied data (References 4 and 5). The MTC value of +4.5 pcm/°F is consistent with the Technical Specification limit of +5.0 pcm/°F at zero power and is conservative based on operating experience at the power level of interest here, even under the most adverse conditions (BOC and no Xenon).

The Doppler Coefficient of -1.18 pcm/°F has been determined to be a bounding value at the power levels of interest in this analysis (Reference 4).

UNIT 1

A Moderator Temperature Coefficient (MTC) of +6.5 pcm/°F has been used in the analysis. This value is consistent with the Technical Specification limit of +7.0 pcm/°F at zero power and is conservative based on operating experience at the power level of interest here, even under the most adverse conditions (BOC and no Xenon).

The Doppler Coefficient of -1.0 pcm/°F used in the analysis is the same as that used in the FSAR for the Loss of Load Analysis. The remaining physics parameters have been selected to envelope those used in the FSAR Loss of Load Analysis.

Steam Generator Pressure

A conservatively low value of 856 psia has been assumed in the analysis for the initial steam generator pressure. The lower the initial steam generator pressure the more conservative are the results of the analysis in terms of RCS peak pressure. This is because with lower initial pressures, it takes longer for the secondary to reach the opening pressure setpoint of the SBCS valves thus allowing more time for the RCS heatup. The initial value of 856 psia has been conservatively estimated as the steam generator pressure that corresponds to a 15% level of tube plugging. A more realistic estimate of this pressure with inclusion of the effects of plugging in the overall heat transfer coefficient of the tubes shows that the initial steam generator pressure should be at least 860 psia.

Steam Header Pressure

The pressure in the steam header (before the turbine) is compared to the SBCS pressure setpoint and the difference or error is fed to the Proportional-Integral-Differentiator (PID) controller that actuates the SBCS valves in their Pressure Modulation mode of operation. The pressure in the steam header has been estimated by assuming a 4 psi pressure drop between the steam generators and the header at the steam flow that corresponds to 27% power. This value envelopes measured pressure drops at the low powers of the analyses in this submittal.

Steam Bypass Control System (SBCS) Setpoints

The RCS pressure excursion during the turbine trip without reactor trip event analyzed here is terminated by the action of the Proportional-Integral-Derivative (PID) master controller (PIC-8810) in the SBCS. This component regulates the action of the five steam dump bypass valves in their pressure modulating mode of operation. The controller setpoints assumed in the analysis are detailed in Table 1.

. System Initial Conditions

Other important initial conditions, such as the primary pressure, pressurizer liquid level, core inlet temperature, steam generator inventory, etc. are assumed to be at the nominal values that correspond to 27% power. The use of these parameters at nominal values is consistent with the goal of determining if the increase in the reactor trip bypass from 15% to 25% power will increase the probability of a small break LOCA. Feedwater flow is assumed to follow its normal ramp to 5% of the nominal flow 3 seconds after the turbine trip. With a relatively large primary power/secondary flow mismatch, a reactor trip on low steam generator water level is anticipated to occur at around 220 seconds after the turbine trip if the operator does not take preventive action to increase the feedwater flow. For the purpose of this analysis, reactor trip on low steam generator level has been ignored.

4. RESULTS

A turbine trip without reactor trip from 27% power has been analyzed with the initial conditions given in Table 1. The analysis sequence of events is shown in Table 2.

Following the turbine trip, the pressure in the steam header increases rapidly from its initial value of 852 psia until it reaches a peak of 920 psia at 11 seconds. The SBCS valves start opening sequentially at 5 seconds. The first two valve groups are fully open at 18 seconds while the third valve group modulates to keep the turbine header pressure at the setpoint value of 900 psia (Figure 1).

As a result of the positive moderator temperature coefficient, the system reaches a new stable condition in which the steam generator pressure and temperature and the primary temperature and core power are higher than initial values. The primary heatup is eventually terminated when the positive reactivity feedback from the coolant temperature is compensated for by the corresponding negative reactivity feedback from the fuel temperature increase (Figures 2 and 3). Because the reactor trip at 25% level is omitted, core power increases until it eventually stabilizes at around 827 MW [905 MW] or 31% [34%] of nominal core power (Figure 4). The pressurizer pressure excursion starts after the turbine trip and reaches a peak of 2365 psia [2383 psia] at 30.5 seconds [38 seconds] (Figure 5). The RCS pressure is turned around when the valves in the 30% steam dump group are able to catch up with the energy accumulated in the primary and eventually stabilizes at around 2300 psia. The RCS temperatures are shown in Figures 6 and 7. The analysis was ended at 200 seconds during the stable new condition described above.

The results of the analysis show that, even with core power at 31% [34%] of nominal, the RCS peak pressure of 2365 psia [2383 psia] is below the PORV opening setpoint and the high pressurizer pressure trip setpoint of 2370 psia [2400 psia]. Therefore, it can be concluded that the probability of a SBLOCA due to a stuck-open PORV is not changed by the proposed setpoint change.

The actual margin between primary peak pressure and PORV opening pressure of 2370 psia [2400 psia] has been evaluated by a reanalysis of the above transient but with no single failure assumed; that is, with no failure of the Pressurizer Pressure Control System. The results of the analysis show that this margin is of the order of 30 psi [40 psia] even with the reactor trip at 25% omitted.

5. CONCLUSION

Turbine trip without reactor trip has been analyzed with all control systems assumed operational and also with single failure assumptions concerning these control systems. The results of the analysis confirm that the proposed change of the reactor trip on turbine trip enable from 15% to 25% power does not result in an increase in the likelihood of PORV opening nor any other hazardous condition.

6. REFERENCES

1. J.H. McFadden, et al, "RETRAN-02, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", EPRI-NP-1850-CCMA, October 1984.

2. Topical Report, "RETRAN Code, Transient Analysis Model Qualifications", Florida Power & Light Company. Fuel Resources Department, Thermal-Hydraulics System Analysis Section. NTH-G-6, July 1985. SER issued on April 19, 1989 in letter of G.C. Lainas to W.F. Conway.

3. Topical Report, "RETRAN Model Qualification, Heat Removal By the Secondary System" Fuel Resources Department, Thermal-Hydraulics System Analysis Section,NTH-TR-01, July 1989. Included in letter from J.H.Goldberg, FPL Executive Vice President, to the U.S. NRC dated October 2, 1989.

4. Letter from E. L. Trapp from Combustion Engineering to H. A. Barth, FPL, "Transmittal of St. Lucie 2 Physics Parameters for Reactor Trip on Turbine Trip Bypass to 25 percent Power", October 4, 1989, F2-CE-R-409, CE Contract No. 17887.

5. Letter to W.L. Parks (FPL) from E.L. Trapp (CE), Subject: St. Lucie Unit 2 Cycle 6 Reload Safety Evaluation (RSE) Report, F2-90-035, July 6, 1990.

TABLE 1

INITIAL CONDITIONS FOR THE ST. LUCIE 1 AND 2 RETRAN MODELS AT 27% POWER

Core Power (MWt)	729
Moderator Temperature Coefficient ($\Delta K/K/°F \times 10^{-5}$)	+4.5 [+6.5]
Doppler Reactivity Coefficient ($\Delta K/K/°F \times 10^{-5}$)	-1.18 [-1.0]
Beta Effective	0.00510
Neutron Generation Lifetime (seconds $\times 10^{-5}$)	2.7
Pressurizer Pressure (psia)	2250
Pressurizer Liquid Level (%NR)	38 [39]
Cold Leg Temperature (°F)	536.6
Coolant Average Temperature (°F)	543.8
RCS Flow (gpm)	363,000 [370,000]
Steam Generator Tube Plugging (%)	15
Steam Generator Pressure (psia)	856
Turbine Header Pressure (psia)	852
Steam Generator Level (%NR)	62.5
Total Feedwater Flow (Million lbm/hr)	2.78
Feedwater Enthalpy (Btu/lbm)	281.6
Total Reactor Coolant Pump Power (MW)	16.5
PORV Opening Setpoint (psia)	2370 [2400]
SBCS Valves (5% & 10% Cap.) Stroke Times (sec)	5. & 10.
SBCS PID:	
Pressure Setpoint (psia)	900
Reset Constant (min)	0.04
Rate Constant (min)	0.01

TABLE 2

ST. LUCIE UNITS 1 AND 2
TURBINE TRIP WITHOUT REACTOR TRIP
AT 27% POWER

SEQUENCE OF EVENTS

Event	Setpoint or Value Reached	Time (sec)
Turbine Trip		0.0
5% SBCS Valve (PCV 8801) starts opening		5.5
10% SBCS Valve (PCV 8802) starts opening		7.5
10% SBCS Valves (PCVs 8803, 8804 & 8805) start opening		10.0
PCV 8801 fully open		10.5
Steam Header Peak Pressure	920 psia	10.5
PCV 8802 fully open		17.5
Pressurizer Peak Pressure	2365 psia [2383 psia]	30.5 [38.0]
PCVs 8803, 8804 & 8805 (30% combined capacity) start modulating at around 20% [22%] combined capacity		70.0 [80.0]
T_{AV} Reaches Asymptotic Value	552.0 °F [553.0 °F]	180.0
Core Power Reaches Asymptotic Value	827 MW [905 MW]	180.0

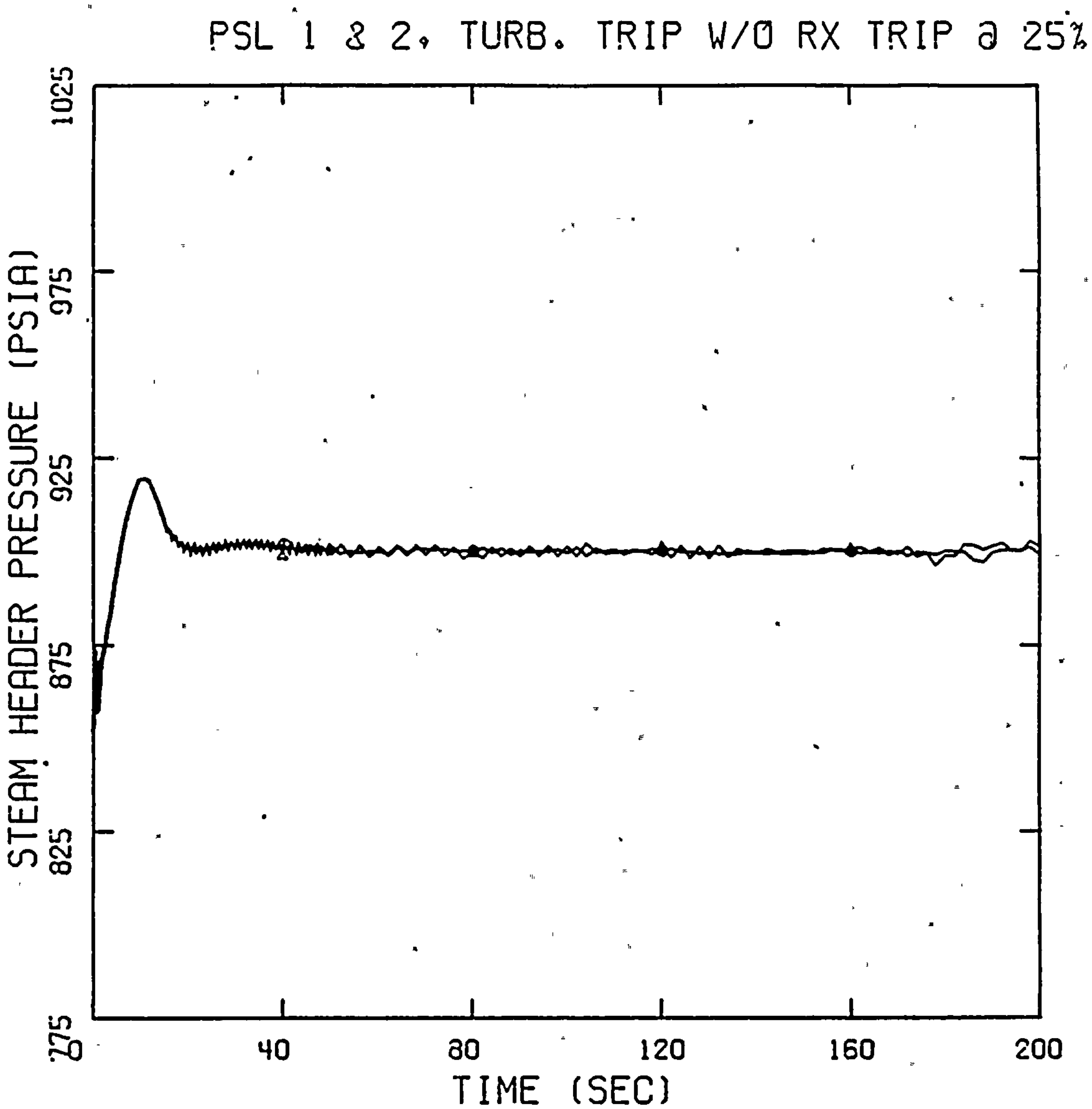

Figure 1. Units 1 & 2, Steam Header Pressure

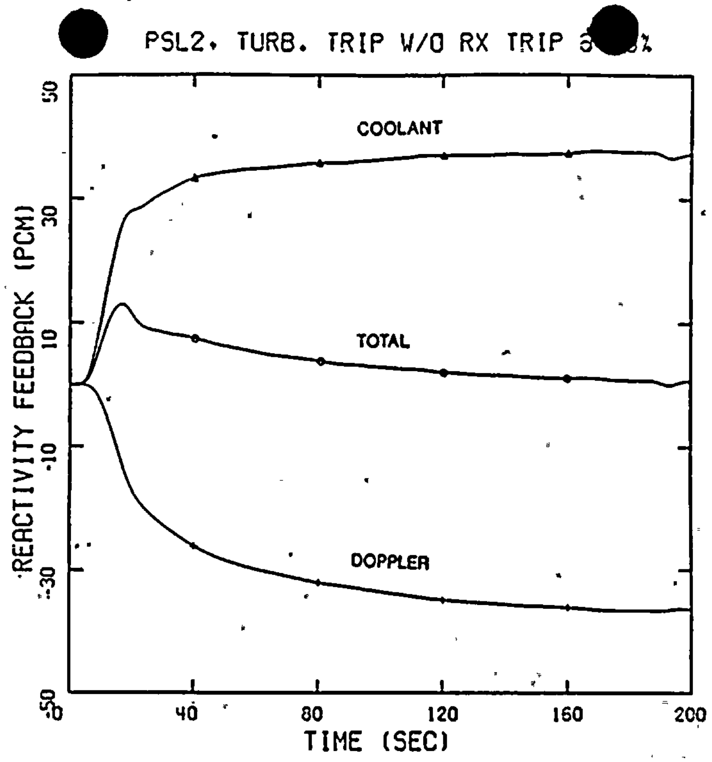

Figure 2. Unit 2, Reactivity Feedback

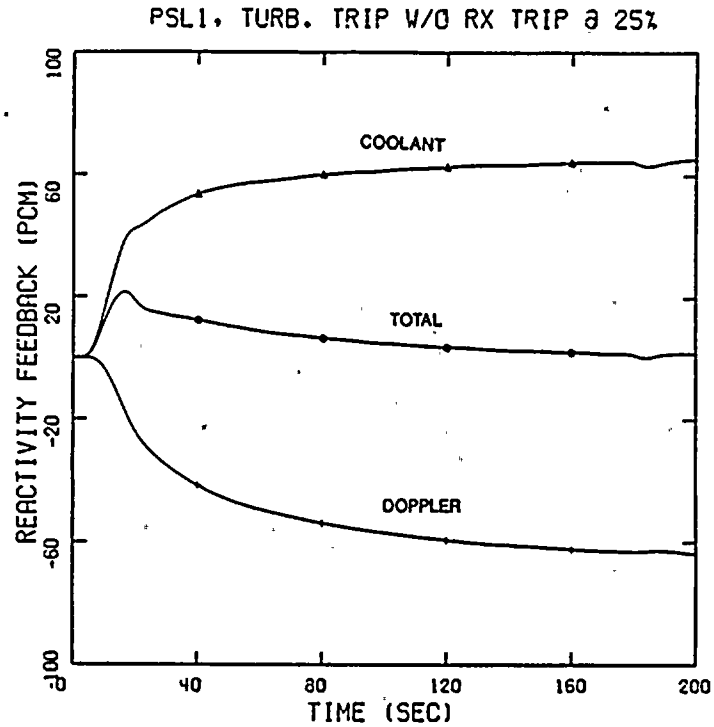

Figure 3. Unit 1, Reactivity Feedback

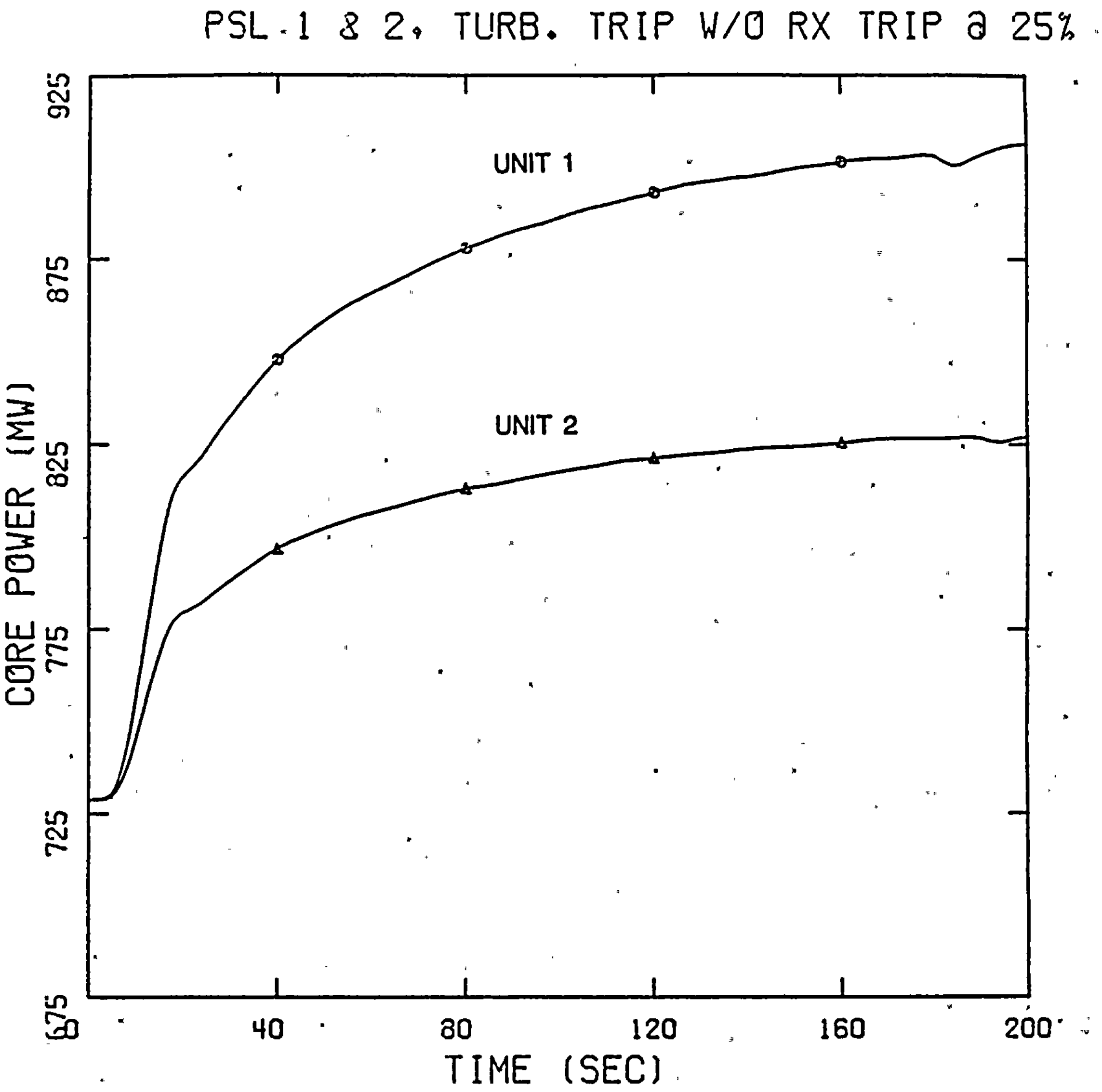

Figure 4. Units 1 & 2, Core Power

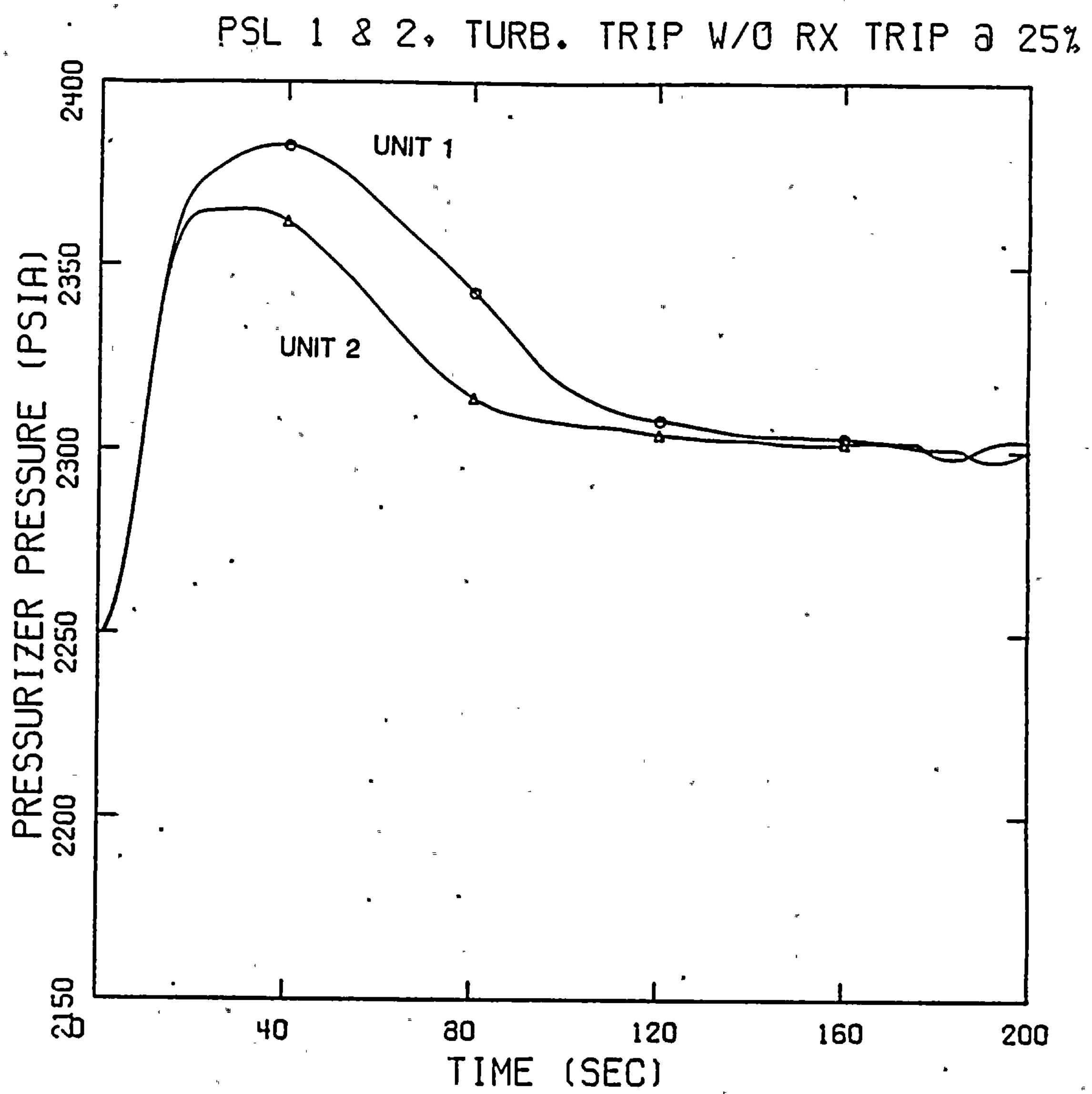

Figure 5. Units 1 & 2, Pressurizer Pressure

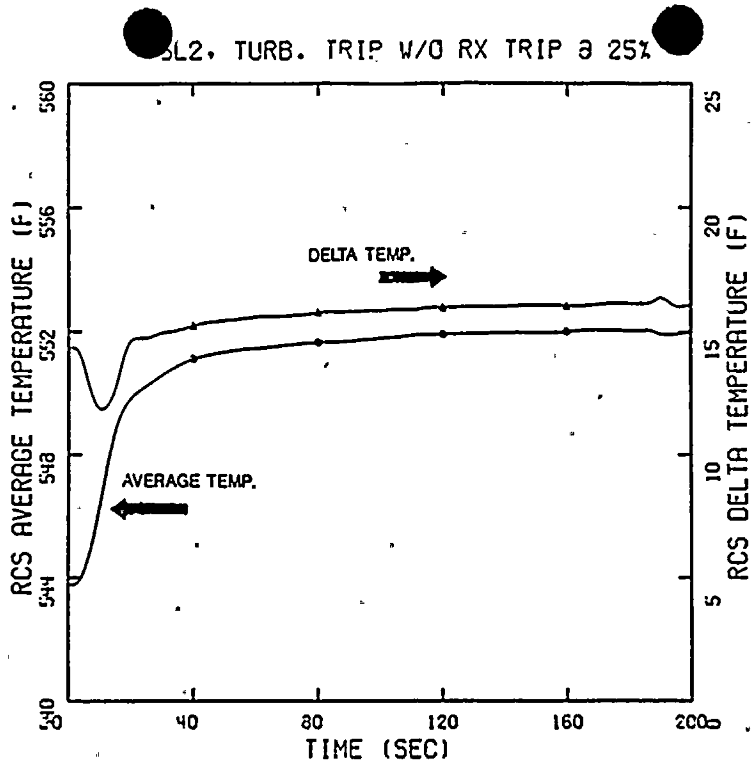

Figure 6. Unit 2, RCS Temperatures

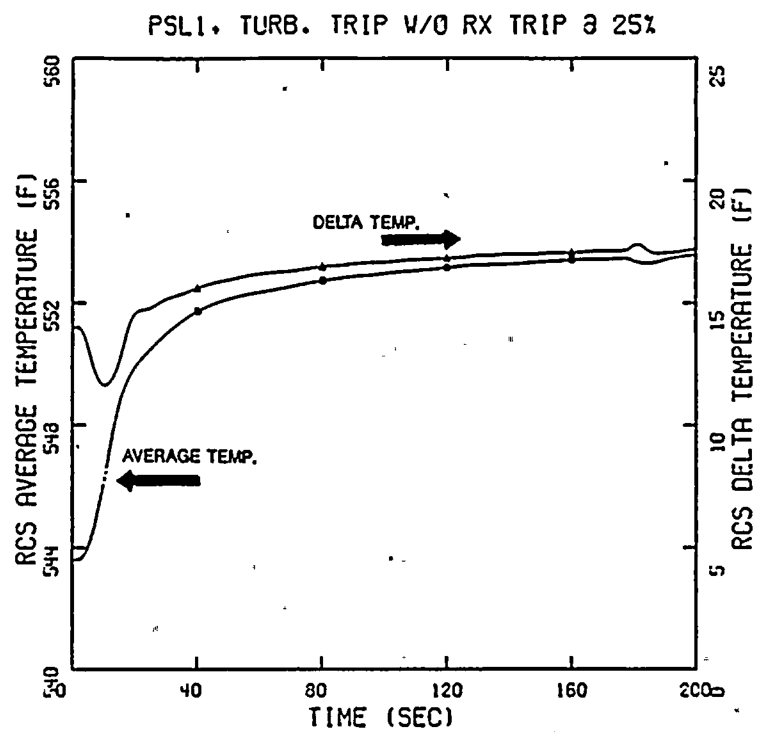

Figure 7. Unit 1, RCS Temperatures

ATTACHMENT 4

DETERMINATION OF NO SIGNIFICANT HAZARDS

The change proposed here extends the reactor trip inhibit on turbine trip from 15% to 25% power level for St. Lucie Units 1 and 2. The evaluation performed in support of this change has determined that when measured against the standard of 10 CFR 50.92, no significant hazardous condition exists.

It has been determined that the operation of St. Lucie Units 1 and 2 with the proposed change does not:

1. **Involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The accidents in the FSAR have been reviewed in terms of a possible increase in their probability of occurrence due to the proposed change. The event of relevance here is a small break LOCA resulting from a stuck-open PORV. A turbine trip without reactor trip at 25% power, which could lead to PORV opening, has been analyzed with the most limiting single failure criterion applicable to the St. Lucie Units 1 and 2: failure of pressurizer spray. The results of the analysis demonstrate that PORV opening does not occur if the Steam Bypass Control System (SBCS) is available. Therefore, the probability of occurrence of a Small Break LOCA (SBLOCA) is not affected by the proposed change. The scenario in which the SBCS is not available has also been analyzed, with the conclusion that the probability of occurrence of a SBLOCA is not increased by the proposed setpoint change.

   The proposed change will not increase the consequences of an accident previously evaluated, since this change only impacts the power range between 15% and 25% where no previously evaluated transient is determined to be limiting.

2. **Create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The proposed change only affects plant operation in the range of powers between 15% and 25%. Outside this range, the plant will continue to operate as before and its safe operation is demonstrated by the analyses included in the FSAR. Within the affected power range, that is between 15% to 25%, no new accident will occur simply due to the proposed setpoint change.

3. Involve a significant reduction in the margin of safety.

The results of the analyses show that implementation of the proposed change does not involve a significant reduction in the margin of safety. The plant continues to operate as before for all power ranges except in the range between 15% and 25% where the reactor trip on turbine trip inhibit feature is extended. Within this range, the effects of the proposed change on the existent safety margins are well within the acceptance criteria for safe operation of the plant.

Based on its compliance with 10 CFR 50.92 criteria, it is concluded that the proposed change meets the requirements for a no significant hazard consideration.