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RAISED SAFETY/RELIEF VALVE SETPOINT
REANALYSIS FOR THE
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
FOR RELOAD NO. 2

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CONTENTS OF THIS REPORT
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1.0 Introduction and Summary

One event that has a significant impact on Boiling Water Reactor (BWR) availability is the spurious opening or failure to reclose of the dual function safety/relief valves. As described in Reference 1, the event from a safety standpoint has a relatively minor effect on the reactor core and reactor coolant pressure boundary. However, the event can result in a significant maintenance outage since the reactor must be shutdown, depressurized, and the valve repaired or replaced before the plant can be returned to service.

The cause of the majority of these spurious openings or failures to reclose of safety/relief valves is excessive leakage around the setpoint pilot valve. Other causes of valve failures have been identified and corrective action has been taken. Operating data demonstrate that an increase in valve simmer margin (the differential pressure between the valve setpoint and normal system operating pressure at the valve) will reduce the probability of valve failure due to pilot leakage.

The Reload 2 licensing supplement (Reference 2) presented the results of the safety analysis for Cycle 3 using the following safety/relief valve (S/RV) groupings and setpoints:

2 @ 1090 psig + 1%
9 @ 1115 psig + 1%

Subsequent calculations performed to address the Commission's multiple safety/relief valve actuations request (Reference 3) indicated that multiple actuations could be minimized, while obtaining the benefit of an increased simmer margin, by using the following revised valve groupings and setpoints:

2 @ 1090 psig + 1%
2 @ 1105 psig + 1%
7 @ 1140 psig + 1%

This document provides the results of a reanalysis of the most limiting thermal and pressurization transients using the above revised S/RV setpoint groupings. The results indicate that the operating limits for 8x8R fuel presented in the Reload 2 licensing submittal (Reference 2) are still applicable. The operating limits for 7x7 and 8x8 fuel decreased by 0.01 between EOC3-2 Gwd/t and EOC3-1 Gwd/t. The operating limit for 8x8 fuel is also reduced by 0.01 between EOC3-1 Gwd/t and EOC3. The analysis also indicates that a 103 psig margin to the ASME vessel code limit of 1375 psig exists for the most severe overpressurization event, an MSIV closure with flux scram at EOC3. These results indicate that additional safety/relief valve reliability is obtained without imposing additional restrictions on plant operation.

2.0 Safety Analysis

2.1 Introduction

The safety analysis for FitzPatrick Reload 2 is provided in Reference 2. The raising of the safety/relief valve setpoints only affects those events which result in valve operation to limit system pressure. The limiting events which require reanalysis are the most severe pressurization transient (generator load rejection with failure of the bypass valve), vessel overpressure protection analysis (closure of all main steam line isolation valves - flux scram) and loss-of-coolant accident (small break). In addition, the capability of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems were reevaluated for the higher safety/relief valve setpoints. The results of the analysis which demonstrate the acceptability of the increased simmer margin are given in Sections 2.2 through 2.5.

All analyses were performed using the same input parameters as documented in Reference 2 with the exception of safety/ relief valve setpoints and capacity, which were as follows:

Valve Group	No. of Valves	Setpoint (psig)	ASME Capacity* at Setpoint (Per Valve, 10^5 lbm/hr)
1	2	1090 psig + 1%	7.986
2	2	1105 psig + 1%	8.091
3	7	1140 psig + 1%	8.343

*Capacities include the 2.3% reduction due to use of Schedule XXS inlet piping.

The use of the maximum S/RV setpoint tolerance of +1% results in valve actuations at 1101 psig (Group 1); 1116 psig (Group 2); and 1151 psig (Group 3) in the analysis. The total capacity of all 11 S/RVs at the lowest setpoint is 84.2% of NBR steam flow.

2.2 Generator Load Rejection With Bypass Failure

This transient produces the most severe reactor isolation event during Cycle 3. Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in a significant loss of generator load. The turbine control valves close rapidly to prevent overspeed of the turbine-generator rotor. This closing, concurrent with the failure of the bypass valve system, causes a sudden reduction in steam flow which results in a nuclear system pressure increase. The pressure increase causes a significant void reduction which yields a pronounced positive void reactivity effect. The net reactivity is sharply positive and causes a rapid increase in neutron flux until the net reactivity is forced negative by a scram initiated from closure of the turbine control valves and by a void increase after the safety/relief valves have automatically opened on high pressure.

The results of the load rejection analyses are given in Table 1 and shown in Figures 1 and 2. The peak vessel and steamline pressure are both approximately 10 psig above the results reported in the Reload 2 licensing submittal, but are still well below the ASME vessel code limit of 1375 psig.

In regard to the fuel thermal margins, the load rejection without bypass event determines the MCPR operating limit from EOC3-2GWd/t to EOC3. The GETAB transient analysis results indicate a decrease in the change in critical power ratio (ΔCPR) such that the operating limits for 7x7 and 8x8 fuel are reduced by 0.01 between EOC3-2GWd/t and EOC3-1 Gwd/t. The operating limit for 8x8 fuel is also reduced by 0.01 between EOC3-1 Gwd/t and EOC3. The analysis also indicates that the operating limits presented in the Reload 2 licensing submittal (Reference 2) for 8x8R fuel are still applicable.

From BOC3 to EOC3-2GWd/t the rod withdrawal error transient determines the operating limit for 7x7 and 8x8 fuel, and the loss of 80°F feedwater heating transient determines the operating limit for 8x8R fuel. These transients are not affected by the revised safety/ relief valve setpoints since the safety/relief valves are not actuated during these transients. Therefore, the operating limits for BOC3 to EOC3-2GWd/t presented in the Reload 2 licensing submittal (Reference 2) are still applicable. The MCPR operating limits for Cycle 3 are summarized in Table 2.

2.3 Vessel Overpressure Protection Analysis

The pressure relief system must prevent excessive overpressurization of the primary system process barrier and the pressure vessel to preclude an uncontrolled release of fission products.

The James A. FitzPatrick Plant pressure relief system includes 11 dual function safety/relief valves located on the main steamlines within the drywell between the reactor vessel and the first isolation valve. These valves provide the capacity to limit nuclear system overpressurization.

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from the consequences of pressure in excess of the vessel design pressure:

(a) A peak allowable pressure of 110% of the vessel design pressure is allowed (1375 psig for a vessel with a design pressure of 1250 psig).

(b) The lowest qualified safety valve setpoint must be at or below vessel design pressure.

(c) The highest safety valve setpoint must not be greater than 105% of vessel design pressure (1313 psig for a 1250 psig vessel).

The James A. FitzPatrick Plant safety/relief valves are set to self-actuate at 1090, 1105 and 1140 psig, satisfying (b) and (c) above. Requirement (a) is evaluated by considering the most severe isolation event with failure of the direct scram trip. The safety/relief valves are assumed to activate in their safety valve mode of operation.

The event which satisfies this specification is the closure of all main steamline isolation valves with indirect (flux) scram. The results of the analysis of this event using the revised safety/relief valve setpoints are given in Table 1 and shown in Figure 3. An abrupt pressure and power rise occur as soon as the reactor is isolated. Reactor shutdown is initiated when the neutron flux reaches the 120% high flux scram setpoint. The safety/relief valves open to limit the pressure rise at the bottom of the vessel to 1272 psig. This response provides a 103 psi margin to the vessel code limit of 1375 psig. Thus, requirement (a) is satisfied and adequate overpressure protection is provided by the pressure relief system.

2.4 Loss-of-Coolant Accident Analyses

The revised safety/relief valve setpoints have no significant effect on the large break LOCA analysis results. This is because the system depressurizes so rapidly from the break that the safety/relief valves are not actuated.

A new analysis of the previously limiting small break was performed. The small break models described in Reference 4 were used in the analysis. The results of the analysis showed a 33°F decrease in the calculated peak cladding temperature (PCT) for the most limiting small break (0.07 ft^2 recirculation suction line break with HPCI failure) from the 1285°F PCT reported in Reference 5.

The small break LOCA PCT is therefore reduced from 1285°F to 1252°F. The reason for this decrease is that with the revised S/RV setpoints the vessel depressurizes faster during the small break LOCA, allowing the Core Spray and LPCI systems to come on earlier, resulting in earlier cladding cooling and a lower PCT.

2.5 HPCI and RCIC Capability

One of the design requirements for the HPCI and RCIC systems is that they be capable of providing design flow at the lowest safety/relief valve setpoint. These systems still meet the design requirement of full flow discharge to the core at 1120 psig with the increase in the lowest safety/relief valve setpoint to 1090 psig + 1%.

3.0 References

1. James A. FitzPatrick Nuclear Plant, Final Safety Analysis Report, Docket 50-333, November 1971.

2. "Supplemental Reload Licensing Submittal for the James A. FitzPatrick Nuclear Power Plant for Reload No. 2," June 1978 (NEDO-24129).

3. Letter, Victor Stello (NRC) to Licensees, "Multiple-Subsequent Actuations of Safety/Relief Valves Following an Isolation Event," dated March 20, 1978.

4. "Loss-of-Coolant Accident Analysis Report for the James A. FitzPatrick Nuclear Power Plant (Lead Plant)," July 1977 (NEDO-21772-2).

5. "Supplemental Licensing Submittal for the James A. FitzPatrick Nuclear Power Plant for Reload-1 Operation Between EOC2-Gwd/t and EOC2," March 1978 (NEDO-21619-1).

TABLE 1
FITZPATRICK CYCLE 3
EVENT DATA SUMMARY

Event	Power (%)	Core Flow (%)	$\hat{\phi}$ (%)	$\hat{Q}/A$ (%)	$\hat{P}_{SL}$ (psig)	$\hat{P}_{V}$ (psig)	Response
Load Rejection - No Bypass, Trip Scram							
From EOC3-2GWd/t to EOC3-1GWd/t							
Using Previous S/RV Setpoints (NEDO-24129)	104	100	320	113	1174	1221	Figure 8 (NEDO-24129)
Using Revised S/RV Setpoints (This Analysis)	104	100	320	112	1185	1232	Figure 1
From EOC3-1GWd/t to EOC3							
Using Previous S/RV Setpoints (NEDO-24129)	104	100	376	115	1178	1225	Figure 12 (NEDO-24129)
Using Revised S/RV Setpoints (This Analysis)	104	100	376	114	1187	1235	Figure 2
MSIV Closure, Flux Scram, EOC3							
Using Previous S/RV Setpoints (NEDO-24129)	104	100	-	-	1217	1264	Figure 16 (NEDO-24129)
Using Revised S/RV Setpoints (This Analysis)	104	100	-	-	1226	1272	Figure 3

$\hat{\phi}$ - Peak Neutron Flux (% Initial)

$\hat{Q}/A$ - Peak Heat Flux (% Initial)

$\hat{P}_{SL}$ - Peak Steamline Pressure

$\hat{P}_{V}$ - Peak Vessel Pressure

TABLE 2
FITZPATRICK CYCLE 3 OPERATING LIMITS
WITH REVISED* S/RV SETPOINTS

Exposure Range	MCPR Operating Limit**		
	7 x 7	8 x 8	8 x 8R
From BOC3 to EOC3-2GWd/t	1.21	1.22	1.20
From EOC3-2GWd/t to EOC3-1GWd/t	1.25	1.33	1.33
From EOC3-1GWd/t to EOC 3	1.30	1.37	1.37

* The valve groupings and setpoints are: 2 @ 1090 psig, 2 @ 1105 psig, and 7 @ 1140 psig.

**The Safety Limit MCPR is 1.07.

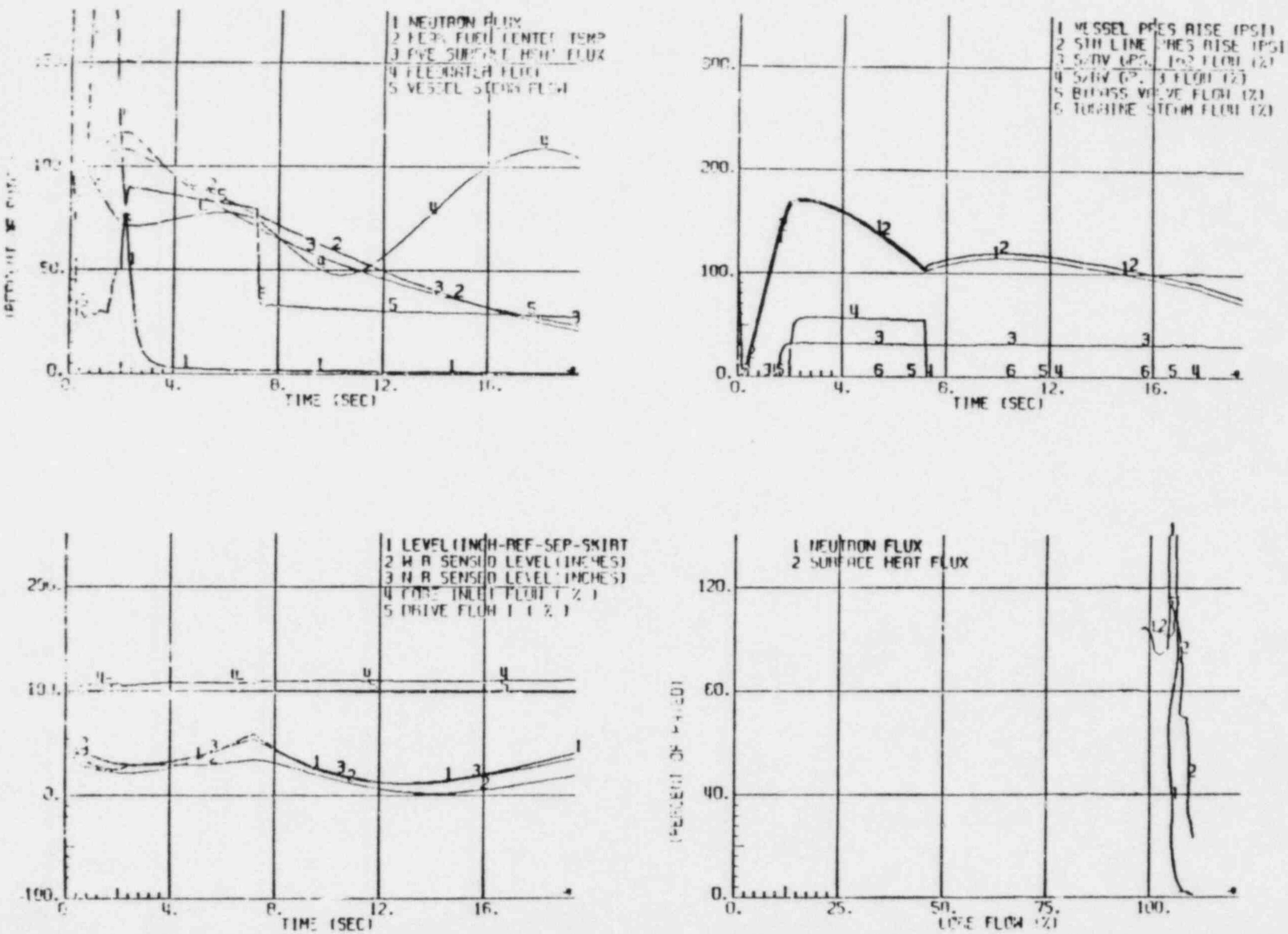

Figure 1. Generator Load Rejection Without Bypass - From EOC3-2GWd/t to EOC3-1 GWd/t

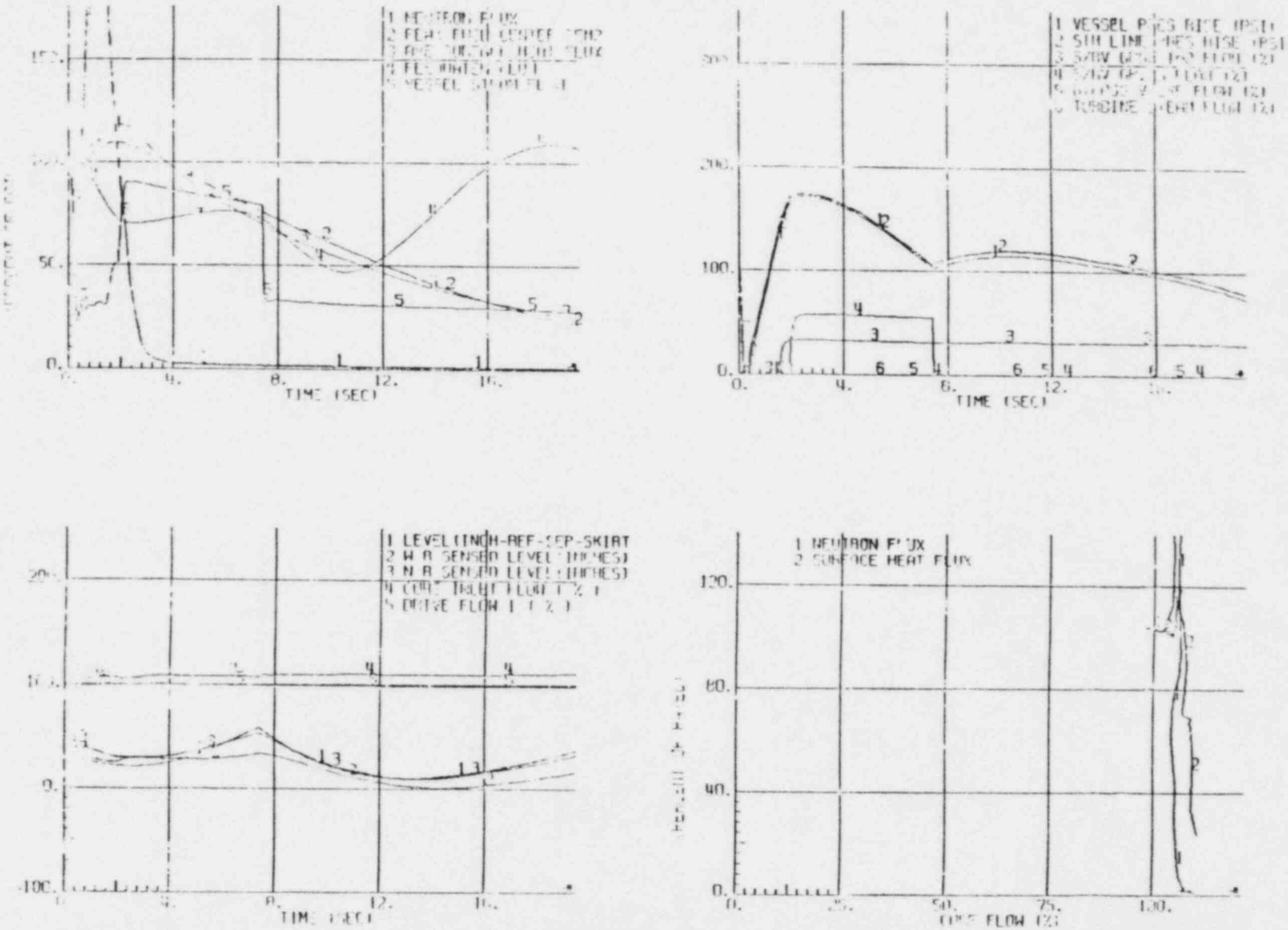

Figure 2. Generator Load Rejection Without Bypass - From EOC3-1 GWd/t to EOC3

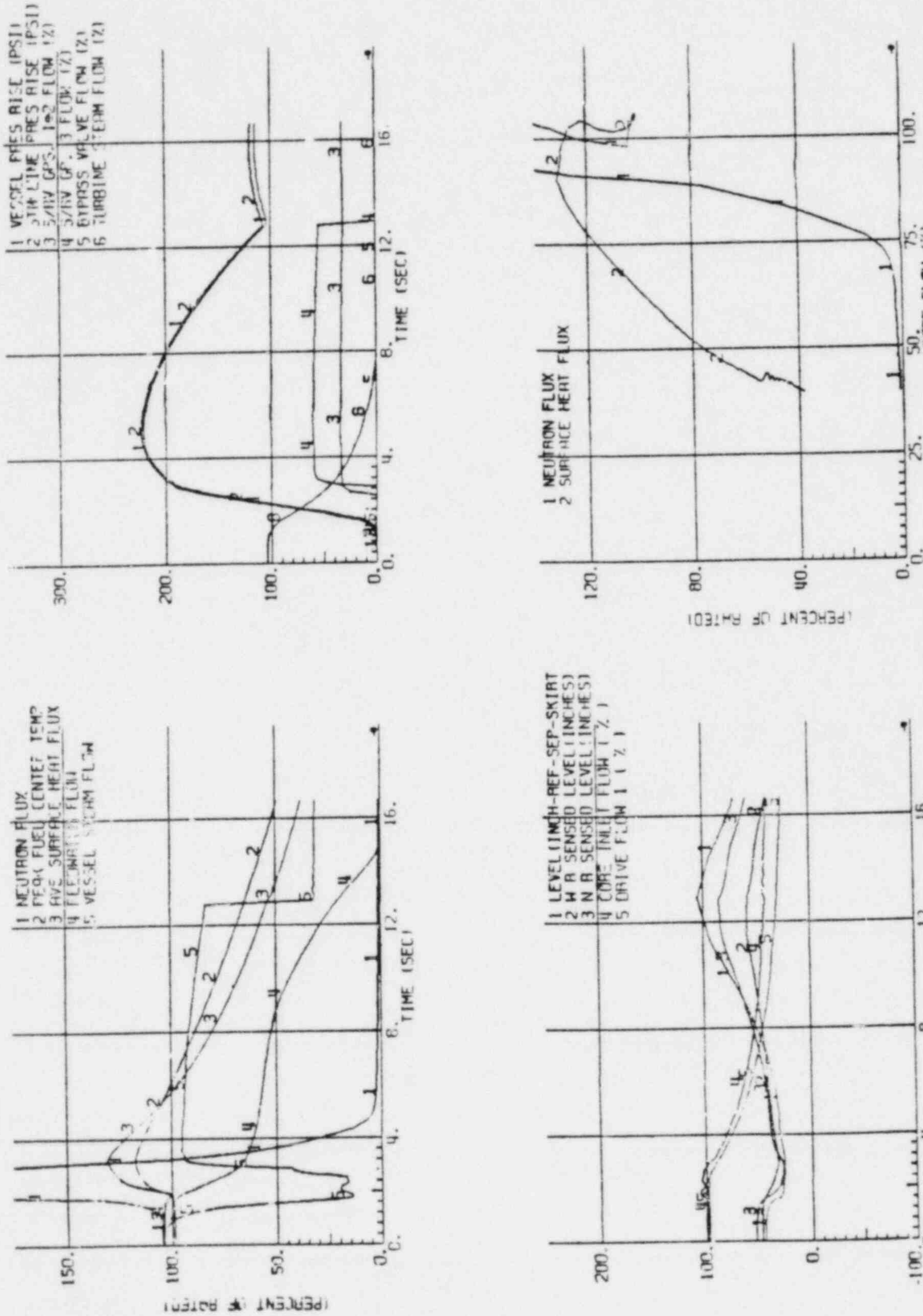

Figure 3. MSIV Closure, Flux Scram, at EOC3

ATTACHMENT A

Power Authority of the State of New York

License No. DPR-59

Docket No. 50-333

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor coolant system pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be $\leq$ 1,045 psig.

   B. Reactor coolant system safety/relief valve nominal settings shall be as follows:

      Safety/Relief Valves

      2 valves at 1090 psig
      2 valves at 1105 psig
      7 valves at 1140 psig

      The allowable setpoint error for each safety/relief valve shall be $\pm$ 1 percent.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575°F for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 = 1,375 psig), and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The analysis in NEDO-24129, "Supplemental Reload Licensing Submittal for the James A. FitzPatrick Nuclear Power Plant for Reload No. 2", June 1978, as amended by NEDO-24129-1, Supplement 1, September 1978, shows that the main steam isolation valve transient, when direct scram is ignored, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. Figure 3 in NEDO-24129-1 presents the curve produced by this analysis. Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal system (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 100 msec.

B. Minimum Critical Power Ratio (MCPR)

During reactor power operation at rated power and flow, the MCPR operating limits shall not be less than those shown below:

FUEL TYPE	MCPR OPERATING LIMIT FOR INCREMENTAL CYCLE 3 CORE AVERAGE EXPOSURE		
	BOC3 to 2GWd/t before EOC3	EOC3-2GWd/t to EOC3-1GWd/t	EOC3-1GWd/t to EOC3
7x7	1.21	1.25	1.30
8x8	1.22	1.33	1.37
8x8R	1.20	1.33	1.37

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Maximum Fraction of Limiting Power Density (MFLPD)

The MFLPD shall be determined daily during reactor power operation at ≥ 25% rated thermal power and the APRM high flux scram and Rod Block trip settings adjusted if necessary as required by Specifications 2.1.A.1.c and 2.1.A.1.d, respectively.

3.1 BASES (cont'd)

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500 < P < 850 psig) to the normal EHC oil pressure of 1,600 psig so that, based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the startup and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis for Cycle 3 (NEDO-24129 and NEDO-24129-1, Supplement 1) for various core exposures are given in Specification 3.1.B.

The ECCS performance analysis assumed reactor operation will be limited to MCPR of 1.18. However, the Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as given in Specification 3.1.B.

Amendment No. ~~14~~, ~~18~~, ~~21~~, ~~30~~, ~~35~~

3.3 and 4.3 BASES (cont'd)

resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (NEDO-24129-1 Figures 1 and 2) with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than the Safety Limit.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on JAFNPP.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 290 msec are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 msec estimated from the scram test results. Approximately 90 msec of each of these intervals result from the sensor and the circuit delay, at this point, the pilot scram valve solenoid de-energizer. Approximately 120 msec later, control rod motion is estimated to actually begin. However, 200 msec is conservatively assumed for this time interval in the transient analysis and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to de-energize the pilot valve scram solenoid is measured during the calibration tests required by Specification 4.1.

The scram times generated at each refueling outage and during operation when compared to scram times generated during pre-operational tests demonstrate that the control rod drive scram function has not deteriorated. In addition, each instant when control rods are scram timed during operation or reactor trips, individual evaluations shall be performed to insure that control rod scram times have not deteriorated.

D. Reactivity Anomalies

During each fuel cycle, excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses anomalous behavior in the excess reactivity may be detected by comparison of

Amendment No. ~~14~~, ~~18~~, ~~21~~, ~~30~~, ~~35~~