ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING UPDATED SRV PERFORMANCE REQUIREMENTS

(JPTS-89-017)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

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 vessel dome 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. At least 9 of the 11 reactor coolant system safety/relief valves shall have opening pressures less than or equal to an upper limit value of 1195 psig.

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 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 limiting vessel overpressure transient event is a main steam isolation valve closure with flux scram. This event was analyzed within NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP", assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to an upper limit value of 1195 psig. The resultant peak vessel pressure for the event was shown to be less than the vessel pressure code limit of 1,375 psig. (See current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event.) The upper limit value of 1195 psig is the SRV opening pressure up to which plant performance has been analyzed, assuming 2 SRVs are inoperable. Therefore, SRV opening pressures below the upper limit (1195 psig) ensure that the ASME Code limit on peak reactor pressure is satisfied.

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.

TABLE 4.2-2 (Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

	Logic System Functional Test		Frequency
1)	Core Spray Subsystem	(7) (9)	Once/6 months
2)	Low Pressure Coolant Injection Subsystem	(7) (9)	Once/6 months
3)	Containment Cooling Subsystem	(9)	Once/6 months
4)	HPCI Subsystem	(7) (9)	Once/6 months
5)	HPCI Subsystem Auto Isolation	(7) (9)	Once/6 months
6)	ADS Subsystem	(7) (9)	Once/6 months
7)	RCIC Subsystem Auto Isolation	(7) (9)	Once/6 months

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

3.5 (cont'd)

D. Automatic Depressurization System (ADS)

1. The ADS shall be operable with at least 5 of the 7 ADS valves operable:

   a. whenever the reactor pressure is greater than 100 psig and irradiated fuel is in the reactor vessel, and

   b. prior to reactor startup from a cold condition.

4.5 (cont'd)

D. Automatic Depressurization System (ADS)

1. Surveillance of the Automatic Depressurization System shall be performed during each operating cycle as follows:

   a. A simulated automatic initiation which opens all pilot valves.

   b. A simulated automatic initiation which is inhibited by the override switches.

3.5 (cont'd)

4.5 (cont'd)

2. If the requirements of 3.5.D.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 100 psig within 24 hr.

3. Low power physics testing and reactor operator training shall be permitted with inoperable ADS components, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or reactor vessel head is removed.

3.5 BASES (cont'd)

C. High Pressure Coolant Injection (HPCI) System

The High Pressure Coolant Injection System is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or Core Spray Systems can protect the core.

The HPCI meets this requirement without the use of a-c electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. Refer to Section 6.5.3 of the FSAR.

Low power physics testing and reactor operator training with inoperable component(s) will be conducted only when the HPCI System is not required, (reactor coolant temperature $\leq$212°F and coolant pressure $\leq$150 psig). If the plant parameters are below the point where the HPCI System is required, physics testing and operator training will not place the plant in an unsafe condition.

Operability of the HPCI System is required only when reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F because core spray and low pressure coolant injection can protect the core for any size pipe break at low pressure.

D. Automatic Depressurization System (ADS)

The relief valves of the ADS are a backup to the HPCI subsystem. They enable the Core Spray or LPCI Systems to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the Core Spray or LPCI Systems. The core spray and/or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad fragmentation and to assure that core geometry remains intact.

The ADS has sufficient excess capacity such that only five of the seven valves are required operable during power operation (see NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP").

Loss of three ADS valves reduces the pressure relieving capacity, and, thus, a 24 hour action to a cold condition with reactor pressures less than 100 psig is specified.

Low power physics testing and reactor operator training with inoperable components will be conducted only when that component or system is not required, (reactor coolant temperature $\leq$212°F and reactor vessel vented or the reactor vessel head removed). With the reactor coolant temperature $\leq$212°F and the Reactor vessel vented or the

3.6 (cont'd)

E. Safety/Relief Valves

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F, the safety mode of at least 9 of 11 safety/relief valves shall be operable. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D.

4.6 (cont'd)

E. Safety/Relief Valves

1. At least 5 of the 11 safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. All valves shall be tested every two operating cycles. The testing shall demonstrate that the 11 safety/relief valves actuate at 1110 psig ±3%.

3.6 (cont'd)

2. If Specification 3.6.E.1 is not met, the reactor shall be placed in a cold condition within 24 hr.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Specification 3.6.E.1 above, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or the reactor vessel head is removed.

4. The provisions of Specification 3.0.D are not applicable.

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

4. Manually open each safety/relief valve while bypassing steam to the condenser and observe a $\geq$10% closure of the turbine bypass valves, to verify that the safety/relief valve has opened. This test shall be performed at least once each operating cycle within the first 12 hours of continuous power operation at a reactor steam dome pressure of $\geq$ 940 psig.

3.6 and 4.6 BASES (cont'd)

E. Safety/Relief Valves

The safety/relief valves (SRVs) have two modes of operation; the safety mode or the relief mode. In the safety mode (or spring mode of operation) the spring loaded pilot valve opens when the steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. The safety mode of operation is required during pressurization transients to ensure vessel pressures do not exceed the reactor coolant pressure safety limit of 1,375 psig.

In the relief mode the spring loaded pilot valve opens when the spring force is overcome by nitrogen pressure which is provided to the valve through a solenoid operated valve. The solenoid operated valve is actuated by the ADS logic system (for those SRVs which are included in the ADS) or manually by the operator from a control switch in the main control room or at the remote ADS panel. Operation of the SRVs in the relief mode for the ADS is discussed in the Bases for Specification 3.5.D.

Experiences in safety/relief valve testing have shown that failure or deterioration of safety/relief valves can be adequately detected if at least 5 of the 11 valves are bench tested once per operating cycle so that all valves are tested every two operating cycles. Furthermore, safety/relief valve testing experience has demonstrated that safety/relief valves which actuate within ±3% of the design pressure setpoint are considered operable (see ANSI/ASME OM-1-1981). The safety bases for a single nominal valve opening pressure of 1110 psig are described in NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP." The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of 1110 psig preserves the safety margins associated with the HPCI and RCIC turbine overspeed systems and the Mark I torus loading analyses. Based on safety/relief valve testing experience and the analysis referenced above, the safety/relief valves are bench tested to demonstrate that in-service opening pressures are within the nominal pressure setpoints ±3% and then the valves are returned to service with opening pressures at the nominal setpoints ±1%. In this manner, valve integrity and the margin to the upper limit value specified in 2.2.1.B are maintained from cycle to cycle.

The analyses with NEDC-31697P also provide the safety basis for which 2 SRVs are permitted inoperable during continuous power operation. With more than 2 SRVs inoperable, the margin to the reactor vessel pressure safety limit is significantly reduced, therefore, the plant must enter a cold condition within 24 hours once more than 2 SRVs are determined to be inoperable. (See reload evaluation for the current cycle).

A manual actuation of each SRV is performed to verify that the valves' are mechanically functional and that no blockage exists in the valve discharge line. Adequate reactor steam dome pressure must be available to perform this test, in accordance with the manufacturer's recommendations, to avoid damaging the valve. Therefore, plant start-up is allowed and sufficient time is provided after the required pressure is achieved (940 psig) to perform this test.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the safety/relief and safety valves are

ATTACHMENT II

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING UPDATED SRV PERFORMANCE REQUIREMENTS

(JPTS-89-017)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

I. DESCRIPTION OF THE PROPOSED CHANGES

This application for an amendment to the James A. FitzPatrick Technical Specifications proposes new Safety/Relief Valve (SRV) performance limits to take credit for the currently installed SRV capacity. Other changes, unassociated with SRV performance, clarify selected portions of the Technical Specifications and correct minor typographical and editorial errors.

A. New SRV Performance Limits

Four changes to the existing SRV performance limits are proposed:

- The first permits continued plant operation with two SRVs out of service. Since 7 of the 11 SRVs at FitzPatrick are also ADS (automatic depressurization system) valves, this reduces the number of ADS valves required to be operable to five. Current specifications permit only one SRV out of service for thirty days.
- Secondly, the setpoint for all eleven SRVs are changed to a single nominal setpoint. Current specifications stagger the setpoints from 1090 to 1140 psig.
- The third change increases the maximum permissible setpoint tolerance from one to three percent.
- Fourth, the Limiting Safety System Setting (LSSS) is defined as 1195 psig.

A comparison of the changes in performance requirements is summarized as follows:

Performance Requirement		Present Limit	New Limit
1.	SRV opening pressure required to prevent overpressurization of the reactor coolant system (TS 2.2.1.B)	+1% of Setpoint	1195 psig
2.	Maximum SRV opening pressure used in other licensing basis analyses (FSAR Chapter 14)	+1% of Setpoint	1195 psig
3.	Nominal SRV Setpoint	2 @ 1090 psig 2 @ 1105 psig 7 @ 1140 psig	11 @ 1110 psig
4.	Setpoint tolerance	±1% of Setpoint	±3% of Setpoint
5.	Number of SRVs and ADS valves assumed to be out-of-service (TS 3.5.D and 3.6.D)	0	2

The specific changes to the FitzPatrick Technical Specifications, which incorporate these new SRV performance limits, are detailed below:

1. Specification 2.2.1.B, page 27; change,

   "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 ± 1 percent."

   to read:

   "At least 9 of the 11 reactor coolant system safety/relief valves shall have opening pressures less than or equal to an upper limit value of 1195 psig."

2. Bases Section 2.2.1.B, page 29; delete the last paragraph (begins with "The numerical distribution...") and change the fourth paragraph (begins with "The current reload analysis...") to read:

   "The limiting vessel overpressure transient event is a main steam isolation valve closure with flux scram. This event was analyzed within NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP," assuming 9 of the 11 SRVs were operable with opening pressures less than or equal to an upper limit value of 1195 psig. The resultant peak vessel pressure for the event was shown to be less than the vessel pressure code limit of 1,375 psig. (See current reload analysis for the reactor response to the main steam isolation valve closure with flux scram event.) The upper limit value of 1195 psig is the SRV opening pressure up to which plant performance has been analyzed, assuming 2 SRVs are inoperable. Therefore, SRV opening pressures below the upper limit (1195 psig) ensure that the ASME Code limit on peak reactor pressure is satisfied."

3. Specification 3.5.D.1, page 119 and 120; replace specification with the following:

   The ADS shall be operable with at least 5 of the 7 ADS valves operable:

   a. whenever the reactor pressure is greater than 100 psig and irradiated fuel is in the reactor vessel, and

b. prior to reactor startup from a cold condition.

4. Specification 3.5.D.3, page 120; delete the cross-reference to action statements 3.5.D.1.a and 3.5.D.1.b and add "ADS." The revised specification reads:

    "Low power physics testing and reactor operator training shall be permitted with inoperable ADS components, provided that reactor coolant temperature is ≤212°F and the reactor vessel is vented or reactor vessel head is removed."

5. Specification 4.5.D.2, page 120. Delete this specification.

6. Basis Section 3.5.D, page 128; change the second paragraph (begins with, "Redundancy has been provided...") to read as follows:

    "The ADS has sufficient excess capacity such that only five of the seven valves are required operable during power operation (see NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP").

    Loss of three ADS valves reduces the pressure relieving capacity, and, thus, a 24 hour action to a cold condition with reactor pressures less than 100 psig is specified."

7. Specification 3.6.E.1, on page 142a; delete the words "Safety and" from the title, change the word "all" to "at least 9 of 11," and delete the phrase "except as specified by Specification 3.6.E.2." The revised specification shall read as follows:

    "During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F, the safety mode of at least 9 of 11 safety/relief valves shall be operable. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D."

8. Specification 4.6.E.1, page 142a; delete the words "Safety and" from the title, change "one half of all" to "5 of the 11," delete the cross-reference to Specification 2.2.B, and add the revised valve actuation setpoints. The revised specification shall read as follows:

    "At least 5 of the 11 safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. All valves shall be tested every two operating cycles. The testing shall demonstrate that the 11 safety/relief valves actuate at 1110 psig ±3%."

9. Specification 3.6.E.2, page 143; delete this specification.

10. Specification 3.6.E.3 , page 143; delete the cross-reference to specification 3.6.E.2 and renumber this specification to be 3.6.E.2

11. Specification 3.6.E.4, page 143; change the cross-reference from "Item B.2" to "specification 3.6.E.1" and renumber this specification to be 3.6.E.3.

12. Bases Section 3.6 and 4.6, page 152; delete the first paragraph (begins with "Experiences in safety valve...") and change the third paragraph (begins with "The safety function is ...") and fourth paragraph (begins with "It is realized that...") to read:

    Experiences in safety/relief valve testing have shown that failure or deterioration of safety/relief valves can be adequately detected if at least 5 of the 11 valves are bench tested once per operating cycle so that all valves are tested every two operating cycles. Furthermore, safety/relief valve testing experience has demonstrated that safety/relief valves which actuate within ±3% of the design pressure setpoint are considered operable (see ANSI/ASME OM-1-1981). The safety bases for a single nominal valve opening pressure of 1110 psig are described in NEDC-31697P, "Updated SRV Performance Requirements for the JAFNPP". The single nominal setpoint is set below the reactor vessel design pressure (1250 psig) per the requirements of Article 9 of the ASME Code - Section III, Nuclear Vessels. The setting of 1110 psig preserves the safety margins associated with the HPCI and RCIC turbine overspeed systems and the Mark I torus loading analyses. Based on safety/relief valve testing experience and the analysis referenced above, the safety/relief valves are bench tested to demonstrate that in-service opening pressures are within the nominal pressure setpoints ±3% and then the valves are returned to service with opening pressures at the nominal setpoints ±1%. In this manner, valve integrity and the margin to the upper limit value specified in 2.2.1.B are maintained from cycle to cycle.

    The analyses with NEDC-31697P also provide the safety basis for which 2 SRVs are permitted inoperable during continuous power operation. With more than 2 SRVs inoperable, the margin to the reactor vessel pressure safety limit is significantly reduced, therefore, the plant must enter a cold condition within 24 hours once more than 2 SRVs are determined to be inoperable. (See reload evaluation for the current cycle).

B. Miscellaneous Administrative Changes

Five miscellaneous changes are provided to clarify terminology, correct typographical errors, remove a surveillance requirement which should have been deleted as part of Amendment 130, to clarify when SRV manual actuation is performed, and to delete a duplicate specification.

1. Terminology Clarifications

    a. Specification 1.2.1, page 27; change the phrase "reactor coolant system pressure" to "reactor vessel dome pressure."

b. Bases Section 3.6.E and 4.6.E, page 152, change the second paragraph to read:

The safety/relief valves (SRVs) have two modes of operation; the safety mode or the relief mode. In the safety mode (or spring mode of operation) the spring loaded pilot valve opens when the steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. The safety mode of operation is required during pressurization transients to ensure vessel pressures do not exceed the reactor coolant pressure safety limit of 1,375 psig.

In the relief mode the spring loaded pilot valve opens when the spring force is overcome by nitrogen pressure which is provided to the valve through a solenoid operated valve. The solenoid operated valve is actuated by the ADS logic system (for those SRVs which are included in the ADS) or manually by the operator from a control switch in the main control room or at the remote ADS panel. Operation of the SRVs in the relief mode for the ADS is discussed in the Bases for Specification 3.5.D.

2. Typographical Corrections

a. Bases Section 1.2 and 2.2, page 29, second paragraph; change the "-" signs to "=" signs.

b. Specification 3.5.D.2, page 120; delete the "," after "100 psig."

3. Amendment 130 Change

a. Specification 4.2.B, Table 4.2-2, page 80; delete item 8, "ADS Relief Valve Bellow Pressure Switch."

4. SRV Manual Actuation Test

a. Specification 4.5.D.1.b, page 119; move this specification to new Section 4.6.E.4 (page 143) and add "This test shall be performed at least once each operating cycle within the first 12 hours of continuous operation at a reactor steam dome pressure of $\geq$ 940 psig.

b. New Specification 3.6.E.4, page 142; add "The provisions of Specification 3.0.D are not applicable."

c. Bases Section 3.6.E and 4.6.E, page 152; add:

"A manual actuation of each SRV is performed to verify that the valves' are mechanically functional and that no blockage exists in the valve discharge line. Adequate reactor steam dome pressure must be available to perform this test, in accordance with the manufacturer's recommendations, to avoid damaging the valve. Therefore, plant start-up is allowed and sufficient time is provided after the required pressure is achieved (940 psig) to perform this test."

5. Duplicate Specification

   a. Specification 4.6.E.4, page 143, delete the following:

      An annual report of safety/relief valve failures and challanges will be sent to the NRC in accordance with Section 6.9.A.2.b.

II PURPOSE OF THE PROPOSED CHANGES

A. New SRV Performance Limits

Existing Specifications 2.2.1.B and 4.6.E require the SRVs to open with staggered setpoints and with a tolerance of 1%. Specification 4.6.E limits plant operation with one SRV out of service to thirty days. Considering the existing SRV capacity, and recent experiences with SRV setpoint drift, these specifications unnecessarily restrict plant operation based on very conservative SRV performance limits. The proposed changes will reduce forced outages and decrease maintenance and surveillance testing costs; without impacting safety or plant performance.

A detailed analysis of these changes has been performed for the Authority by the General Electric Company. The results of these analyses are summarized in a report entitled "Updated SRV Performance Requirements for the James A. FitzPatrick Nuclear Power Plant" (NEDC-31697P). (Since this report contains proprietary information, copies are being transmitted under a separate cover.) NEDC-31697P predicts plant response assuming that the following new SRV performance limits were adopted:

- relaxation of the $\pm$ 1% nominal valve nameplate setpoint tolerance to $\pm$ 3%,
- operation with any two SRVs or ADS valves inoperable,
- setting all 11 SRVs at a single nominal nameplate setpoint, and
- the Limiting Safety System Setting for the pressure relief system is defined to be 1195 psig.

NEDC-31697P demonstrates that sufficient margin still exists in the reactor vessel overpressure protection, fuel thermal limits, and torus loading analyses if these changes are instituted.

Single Nominal SRV Setpoint

The adoption of a single setpoint for all eleven SRVs will reduce the quantity of spare SRV top-works that must be kept on hand.

Consistent with the Mark I Containment Short Term Program initiatives, Amendment No. 43 (Reference 14) implemented staggered SRV setpoints which limit the number of valves which could experience consecutive actuation following an isolation transient. The Authority has since performed a Mark I Containment Long Term Program assessment (References 15, 16 and 17) which demonstrated that the allowable containment loads are not exceeded due to multiple SRV actuations. A single nominal setting of 1110 psig is selected to preserve the safety margins assumed in the containment loading calculations.

Setpoint Tolerance

Operating experience at the Fitzpatrick plant and at other BWRs has shown that SRV setpoint drift exceeds the setpoint tolerance (See LERs 85-009, 87-004, 88-004, and 88-010, References 1, 2, 3, and 4 respectively). Implementing a 3% setpoint tolerance will lessen the number of valve refurbishments, minimize the number of valves requiring confirmatory testing, and reduce the quantity of reportable events.

Two SRVs Out of Service

The excess installed SRV capacity permits two ADS valves or SRVs to be inoperable during continuous power operations. This will reduce the number of forced outages due to valve inoperability.

Limiting Safety System Setting (LSSS)

Establishing the LSSS as the upper limit opening pressure of 1195 psig is consistent with Standard Technical Specifications and the definition of a Limiting Safety System Setting (Specification 1.H).

B. Miscellaneous Changes

1. Terminology Corrections

   a. This change more clearly specifies where in the reactor coolant system the pressure safety limit of 1325 psig should be measured. Use of the vessel steam dome pressure indicator is consistent with Bases Section 1.2.

   b. This change more clearly defines the methods of SRV actuation. The terminology changes are consistent with the revised wording of Specification 3.6.E.1 and the ADS Bases section.

2. Typographical Corrections

   a. The change to Bases Section 1.2 and 2.2 and Specification 3.5.D.2 correct typographical errors.

3. Amendment 130 Change

   a. The change to Specification 4.2.B deletes the requirement to perform logic functional testing on the ADS bellows pressure switch. This change was inadvertently omitted from Amendment 130 (Reference 11).

4. SRV Manual Actuation Test

   a. Relocates the SRV manual actuation test to the proper Technical Specification Section.

   b. This change clarifies that manual actuation of the SRVs must be performed within 12 hours of achieving the required test pressure of $\geq$ 940 psig. This is consistent with plant and industry practices and has been requested by the NRC Resident Inspector.

   c. Specification 3.0.D and 4.0.D require the successful completion of all surveillance testing prior to plant start-up. This change is in accordance with Standard Technical Specifications and eliminates a literal inconsistency within the Technical Specifications.

5. Duplicate Specification

   a. Specification 4.6.E.4 is deleted because it is redundant to Specification 6.9.A.2.b. Reporting requirements are not surveillance tests and are properly located in Section 6 of the Technical Specifications.

III. IMPACT OF THE PROPOSED CHANGES

A. New SRV Performance Requirements

NEDC-31697P (Reference 5) considered the affects of these changes on eight plant performance issues. The paragraphs below summarize the results of this analysis. Refer to NEDC-31697P for complete details of each analysis.

- Vessel Overpressure Limits: An upper limit opening pressure of 1195 psig results in a 50 psi margin to the ASME Code upset reactor vessel pressure limit of 1375 psig.

- Fuel Thermal Limits: The revised SRV performance requirements have no impact on fuel thermal limits.

- LOCA/ECCS Performance: The peak cladding temperatures of the ECCS/LOCA analysis are insensitive to SRV opening pressure increases. Operation with 2 SRVs or ADS valves out-of-service has an insignificant impact on ECCS/LOCA performance.

- HPCI/RCIC Operability: The revised SRV performance requirements have an insignificant impact on HPCI/RCIC performance. However, the margin to the 125% mechanical overspeed trip for the HPCI and RCIC turbines is reduced by SRV opening pressure increases. The selection of a nominal setting of 1110 psig preserves the turbine overspeed margin by limiting the required turbine speed to 101% of nameplate rating.

- Containment Response and Integrity: The revised SRV performance requirements have no impact on the calculated peak containment pressures and temperatures. An increase in SRV opening pressures to 1195 psig and the resultant increase in SRV discharge loads do not exceed containment structure stress allowables for the limiting load combinations. A nominal setting of 1110 psig is selected to preserve the safety margins included in the Mark I Plant Unique Load Definition Report.

- SRV Simmer Margin: The selection of a 1110 psig single setpoint provides a 110 psi simmer margin and does not increase the occurrence of pilot valve leakage as compared to the current nominal settings (see Section 5.2 of NEDC-31697P). Therefore, the probability of a stuck-open relief valve is not increased.

- 10CFR50 Appendix R - Alternate Shutdown Capability: Post-fire shutdown capability remains within 10CFR50 Appendix R limitations. The revised SRV performance requirements increases the previously analyzed duration of fuel uncovery by 46 seconds. The resultant peak cladding temperature remains below the temperature at which cladding perforations are expected.

- Setpoint Drift to Minus 3%: A revised tolerance of 3% permits setpoint drift down to 1077 psig. This value is 32 psi above the high pressure scram setpoint and provides a sufficient cushion above normal reactor operating pressures.

NEDC-31697P concludes that the changes in the pressure relief system performance requirements do not have a significant safety impact on vessel overpressure margin, fuel thermal limits, LOCA/ECCS performance, HPCI/RCIC operability, containment response, or containment integrity. Furthermore, the performance changes have an insignificant impact on Alternate Shutdown System (10CFR50 Appendix R) performance, simmer margin, and downward setpoint drift.

These new performance limits are primarily administrative changes. The setpoint tolerance of ± 3% and the upper limit value (Limiting Safety System Setting) of 1195 psig are changes reflecting an ASME testing criterion change (Reference 10) and changes to design basis analyses.

Actual physical changes to the plant are minimal. The physical changes are continuous operation with 2 SRVs/ADS valves out-of-service and revised setpoints to 1110 psig.

B. Miscellaneous Administrative Change

These changes are purely administrative in nature. They do not involve a plant modification; nor do they impact any procedural or administrative controls.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated. A bounding analysis [NEDC-31697P, "Updated SRV Performance Requirements for the James A. Fitzpatrick Nuclear Power Plant"] of the revised SRV performance requirements considered plant operation with 9 of 11 SRVs operable and with a common valve actuation pressure of 1195 psig. The analysis demonstrates that a 50 psi margin exists between the maximum anticipated pressure and the American Society of Mechanical Engineers (ASME) Code upset reactor vessel pressure limit of 1375 psig. The analyses of NEDC-31697P also demonstrate that the new SRV performance limits have no significant impact on thermal limits, ECCS/LOCA performance, HPCI/RCIC operability, containment response, containment integrity, or 10CFR50 Appendix R alternate shutdown capability. The analyses also considered simmer margin and downward setpoint drift.

   The five miscellaneous changes clarify terminology, correct typographical errors, remove a surveillance requirement which should have been deleted as part of Amendment 130, clarify when SRV manual actuation is performed, and delete a duplicate specification. These changes are purely administrative in nature and, as such, do not impact previously evaluated accidents or equipment malfunctions.

2. create the possibility of a new or different kind of accident from those previously evaluated. The new SRV performance limits are primarily administrative changes. The only physical changes involve recalibration of SRV setpoints and operation with 2 SRVs/ADS valves out-of service. The operation and function of the pressure relief system are unaffected. No new failure modes are introduced.

   The proposed miscellaneous changes are purely administrative in nature and, as such, do not create the possibility of an accident or malfunction.

3. involve a significant reduction in the margin of safety. The new SRV performance limits slightly reduce the existing margin to vessel overpressure and the margin to the 125% mechanical overspeed trip for the HPCI and RCIC turbines. However, the reduction in the overpressure margin is insignificant (approximately 25 psi) and the plant's response to transients and accidents remains well within the limits established in Section III, Division I of the ASME Boiler & Pressure Vessel Code and the regulatory limits established in General Design Criteria (GDC) 15, Standard Review Plan Section 5.2.2, and FSAR Section 4.4. The reduction in turbine overspeed margin is negligible (less than 1%), because it is within the allowable tolerance of the trip settings.

The proposed miscellaneous changes are purely administrative in nature and do not involve a reduction in safety margin.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not impact the ALARA or Fire Protection Programs at FitzPatrick, nor will the changes
impact the environment.

VI. CONCLUSION

The changes, as proposed, do not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

a. will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

b. will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report;

c. will not reduce the margin of safety as defined in the basis for any technical specification; and

d. involves no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. Licensee Event Report 85-009, Main Steam Safety Relief Valves Found Out of Tolerance During Test.

2. Licensee Event Report 87-004, Main Steam Safety Relief Valves Found Out of Tolerance.

3. Licensee Event Report 88-004, Reactor Safety/Relief Valve Setpoint Drift.

4. Licensee Event Report 88-010, Reactor Safety/Relief Valve Setpoint Drift.

5. NEDC-31697P, Updated SRV Performance Requirements for the James A. Fitzpatrick Nuclear Power Plant, April 1989.

6. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report, Section 4.4 "Pressure Relief System," Section 4.7 "Reactor Core Isolation Cooling System," Section 6.4 "High Pressure Coolant Injection System," and Section 14 "Safety Analyses."

7. USAEC "Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated November 20, 1972.

8. USAEC "Supplement 1 to the Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated February 1, 1973.

9. USAEC "Supplement 2 to the Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated October 4, 1974.

10. ANSI/ASME OM-1-1981, Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices.

11. Amendment 130 to the James A. Fitzpatrick Operating License, May 31, 1989.

12. HPCI Turbine Instruction Manual, Terry Steam Turbine Company, VPF # 2300-61

13. RCIC Turbine Instruction Manual, Terry Steam Turbine Company, VPF # 2059-49-2

14. Amendment 43 to the James A. Fitzpatrick Operating License, November 22, 1978.

15. "Plant Unique Analysis Report of the Torus Suppression Chamber for JAFNPP," Teledyne Engineering Services, TR-5321-1, Revision 1, September 1984.

16. "Plant Unique Analysis Report of the Torus Attached Piping for JAFNPP," Teledyne Engineering Services, TR-5321-2, Revision 1, November 1984.

17. NRC Letter JAF-84-364, dated December 12, 1984, "Post Implementation Audit Review of Unique Analysis Report for Mark I Containment Long Term Program - Program Found Acceptable."

18. ASME Boiler & Pressure Vessel Code, Section III - Rules for Construction of Nuclear Vessels, 1965 Edition with Addenda through Winter 1966.