ATTACHMENT I to JPN-93-050

PROPOSED TECHNICAL SPECIFICATION CHANGES

ELIMINATION OF THE MAIN STEAM ISOLATION VALVE CLOSURE FUNCTION AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR

JPTS-90-008

New York Power Authority

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

3.1 BASES (cont'd)

subchannel. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, main steam isolation valve (MSIV) closure, and generator load rejection, turbine stop valve closure are discussed in Sections 2.1 and 2.2.

Instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the Core and Containment Cooling Systems (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A Reactor Mode Switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (high flux in startup or refuel) System provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The IRM System provides protection against short reactor periods in these ranges.

3.1 BASES (cont'd)

The Control Rod Drive Scram System is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. Each scram discharge instrument volume accommodates in excess of 34 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches 34.5 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR).

The IRM high flux and APRM $\leq$15% power scrams provide adequate coverage in the startup and intermediate range. Thus, the IRM and APRM systems are required to be operable in the refuel and startup/hot standby modes. The APRM $\leq$120% power and flow referenced scrams provide required protection in the power range (reference FSAR Section 7.5.7). The power range is covered only by the APRMs. Thus, the IRM system is not required in the run mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below 217 psig turbine first stage pressure (30 percent of rated), the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting[1]	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
2	APRM Downscale	≥ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤ 1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤ 2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥ 177 in. above TAF	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	≤ 34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ 10% valve closure			X(5)	8 Instrument Channels	A

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TABLE 3.1-1 (cont'd)
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1- (cont'd)

C. High Flux IRM.

D. Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted.

E. APRM 15% Power Trip.

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.

11. See Section 2.1.A.1.

12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit specified in the Core Operating Limits Report.

13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained as specified in the Core Operating Limits Report.

14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.

TABLE 4.1-1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TEST
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/month.(1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Once/month.
Turbine First Stage Pressure Permissive	B	Trip Channel and Alarm (4)	Once/month.(1)(8)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/month.(1)

NOTES FOR TABLE 4.1-1

1. Initially once every month until acceptable failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of JAFNPP.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the instrument channels.

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	Every refueling outage
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	B	Standard Pressure Source	Note (6)
High Drywell Pressure	B	Standard Pressure Source	Note (6)
Reactor Low Water Level	B	Standard Pressure Source	Note (6)
High Water Level in Scram Discharge Instrument Volume	A	Water Column, Note (5)	Once/operating cycle, Note (5)
High Water Level in Scram Discharge Instrument Volume	B	Standard Pressure Source	Every 3 months
Main Steam Line Isolation Valve Closure	A	Note (4)	Note (4)
Turbine First Stage Pressure Permissive	B	Standard Pressure Source	Note (6)

TABLE 4.1-2 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once/operating cycle
Turbine Stop Valve Closure	A	Note (4)	Note (4)

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.

2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.

3. Deleted

4. Actuation of these switches by normal means will be performed during the refueling outages.

5. Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

6. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

3.2 BASES (cont'd)

High radiation monitors in the area of the main steam lines have been provided to detect gross fuel failure as in the control rod drop accident. A trip setting of 3 times normal full-power background is established to close the main steam line drain valves, the recirculation loop sample valves, the mechanical vacuum pump isolation valves, and trip the pumps, to limit fission product release. For changes in the Hydrogen Water Chemistry hydrogen injection rate, the trip setpoint may be adjusted based on a calculated value of the expected radiation level. Hydrogen addition will result in an increase in the N-16 carryover in the main steam.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drops below 825 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of approximately 300 percent of design flow for this high flow or 40°F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip settings of approximately 300 percent for high flow or 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the Safety Limit. The trip

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both trip Systems	Actions (2)
2 (6)	Reactor Low Water Level	≥ 177 in. above TAF	4 Instrument Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Instrument Channels	D
2	Reactor Low-Low-Low Water Level	≥ 18 in. above the TAF	4 Instrument Channels	A
2 (6)	High Drywell Pressure	≤ 2.7 psig	4 Instrument Channels	A
2	High Radiation Main Steam Line Tunnel	≤ 3 x Normal Rated Full Power Background	4 Instrument Channels	E
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4 Instrument Channels	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Instrument Channels	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4 Instrument Channels	B
4	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	8 Instrument Channels	C
2	Low Condenser Vacuum Closes MSIV's	≥ 8" Hg. Vac (7)(8)	4 Instrument Channels	B

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TABLE 3.2-1 (Cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate shutdown cooling.

   E. Isolate the main steam line drain valves, the recirculation loop sample valves, and the mechanical vacuum pumps, within eight hours.

3. Deleted

4. Deleted

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in run mode (interlocked with Mode Switch).

8. Bypassed when mode switch is not in run mode and turbine stop valves are closed.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (8)	Instrument Functional Test	Calibration Frequency	Instrument Check (4)
1)	Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2)	Reactor Low-Low-Low Water Level	(1)(5)	(15)	Once/day
3)	Main Steam High Temp.	(1)(5)	(15)	Once/day
4)	Main Steam High Flow	(1)(5)	(15)	Once/day
5)	Main Steam Low Pressure	(1)(5)	(15)	Once/day
6)	Reactor Water Cleanup High Temp.	(1)	Once/3 months	None
7)	Condenser Low Vacuum	(1)(5)	(15)	Once/day
8)	Main Steam Line High Radiation	(1)(5)	(11)	Once/day

	Logic System Functional Test (7) (9)	Frequency
1)	Main Steam Line Isolation valves Main Steam Line Drain Valves Reactor Water Sample Valves	Once/6 months
2)	RHR - Isolation Valve Control Shutdown Cooling Valves	Once/6 months
3)	Reactor Water Cleanup Isolation	Once/6 months
4)	Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves	Once/6 months
5)	Standby Gas Treatment System Reactor Building Isolation	Once/6 months

NOTE: See notes following Table 4.2-5.

Amendment No. 37, 89, 186, 181, 182

NOTES FOR TABLES 4.2-1 THROUGH 4.2-5

1. Initially once every month until acceptance failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instruments operate in a environment similar to that of JAFNPP.

2. Functional tests are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed within seven (7) days prior to each startup.

3. Calibrations are not required when these instruments are not required to be operable or are tripped. Calibration tests shall be performed within seven (7) days prior to each startup or prior to a pre-planned shutdown.

4. Instrument checks are not required when these instruments are not required to be operable or are tripped.

5. This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

6. These instrument channels will be calibrated using simulated electrical signals once every three months.

7. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

8. Reactor low water level, and high drywell pressure are not included on Table 4.2-1 since they are listed on Table 4.1-2.

9. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

10. At least one (1) Main Stack Dilution Fan is required to be in operation in order to isokinetically sample the Main Stack.

11. Perform a calibration once per operating cycle using a radiation source. Perform an instrument channel alignment once every 3 months using the built-in current source.

12. (Deleted)

13. Calibration and instrument check surveillance for SRM and IRM Instruments are as specified in Tables 4.1-1, 4.1-2, 4.2-3.

14. Functional test is performed once each operating cycle.

15. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

(f) Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

ATTACHMENT II to JPN-93-050

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES

ELIMINATION OF THE MAIN STEAM ISOLATION VALVE CLOSURE FUNCTION AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR

JPTS-90-008

New York Power Authority

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

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES ELIMINATION OF THE MAIN STEAM ISOLATION VALVE CLOSURE FUNCTION AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes to the James A. Fitzpatrick Technical Specification will eliminate the scram and Main Steam Line Isolation Valve (MSIV) closure functions associated with the Main Steam Line Radiation Monitors (MSLRM). The specific changes are described below:

Page 33, Bases 3.1

Delete the following paragraph:

> "High radiation levels in the main steam line tunnel, above levels due to the nitrogen and oxygen radioactivity, are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector offgas monitors which cause an isolation of the main condenser offgas line. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by a factor of approximately 5 at the maximum hydrogen addition rate as indicated in note 16, Table 3.1-1. The scram setpoint will be reset to three times the projected background radiation level prior to performance of the test. The setpoint will be restored to normal following completion of the hydrogen addition test."

Move the start of the last paragraph to page 34.

Page 34, Bases 3.1

Revise to accommodate a redistribution of text from page 33.

Page 41a, Table 3.1-1

Delete the Main Steam Line High Radiation scram requirement.

Page 43, Notes of Table 3.1-1

Delete Note 16 and the footnote regarding the proposed hydrogen addition test.

Page 45, Table 4.1-1

Delete the Main Steam Line High Radiation functional test requirement.

Page 46, Table 4.1-2

Delete the Main Steam Line High Radiation calibration requirement.

Page 47, Table 4.1-2 (cont'd)

Delete Note 3 regarding the calibration of the Main Steam Line High Radiation instrument channel.

Page 57, Bases 3.2

Replace the following paragraph:

> "High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.1.2 FSAR. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by approximately a factor of 5 at the peak hydrogen concentration as indicated in note 16, Table 3.1-1. With the hydrogen addition, the fission product release would still be well within the 10 CFR 100 guidelines in the event of a control rod drop accident."

with:

> "High radiation monitors in the area of the main steam lines have been provided to detect gross fuel failure as in the control rod drop accident. A trip setting of 3 times normal full-power background is established to close the main steam line drain valves, the recirculation loop sample valves, and the mechanical vacuum pump isolation valves, and trip the pumps, to limit fission product release. For changes in the Hydrogen Water Chemistry hydrogen injection rate, the trip setpoint may be adjusted based on a calculated value of the radiation level expected. Hydrogen addition will result in an increase in the N-16 carryover in the main steam."

Page 64, Table 3.2-1

Delete the note "(9)" designation after trip level setting for the High Radiation Main Steam Line Tunnel.

Replace action note "B" for the High Radiation Main Steam Line Tunnel with "E".

Page 65, Table 3.2-1 (cont'd)

Add Note 2.E.

> "Isolate the main steam line drain valves, the recirculation loop sample valves, and the mechanical vacuum pumps, within eight hours."

Delete Note 9 that reads:

> "The trip level setpoint will be maintained at < 3 times normal rated full power background. See note 16 to Table 3.1-1 for re-setting trip level setpoint just prior to and following the Hydrogen Addition Test."

Page 78, Table 4.2-1

Add the following testing requirements for the "Main Steam Line - High Radiation" Instrument Channel:

Instrument Functional Test:	"(1)(5)"
Calibration Frequency:	"(11)"
Instrument Check:	"Once/day"

The proposed calibration frequency for the MSLRM is the same as currently specified in Table 4.1-2 of the Technical Specifications. The proposed monthly frequency for the instrument functional test is consistent with NUREG 0123, Rev. 3, "Standard Technical Specifications for General Electric Boiling Water Reactors," and with the other PCIS trip functions currently in Table 4.2-1.

Page 84, Notes for Table 4.2-1 through 4.2-5

Replace Note 8:

> "Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2-1 since they are tested on Table 4.1-2."

with:

> "Reactor low water level and high drywell pressure are not included on Table 4.2-1 since they are listed on Table 4.1-2."

Replace Note 11:

> "Uses same instrumentation as Main Steam Line High Radiation. See Table 4.1-2."

with:

> "Perform a calibration once per operating cycle using a radiation source. Perform an instrument channel alignment once every 3 months using the built-in current source."

Appendix B (Radiological Effluent Technical Specification), Page 39, Notes for Table 3.10-2

Replace the references to "Tables 4.1-1 and 4.1-2 respectively" in note (g) with "Table 4.2-1."

II. PURPOSE OF THE PROPOSED CHANGES

The Main Steam Line Radiation Monitor (MSLRM) system consists of four redundant radiation detectors located external to the main steam lines outside of primary containment. The monitors are designed to detect a gross release of fission products indicative of fuel failure. The MSLRM currently provides readout, alarm, and trip functions upon detection of excessive radiation levels. A trip initiates a reactor scram, isolates the mechanical vacuum pumps, and initiates a Group I primary containment isolation signal for the Main Steam Isolation Valves (MSIV), main steam line drain valves, and recirculation loop sample valves.

Elimination of the MSLRM scram and MSIV closure functions was recognized as a generic improvement by the BWR Owners Group, who submitted a safety evaluation justifying removal of these functions to the NRC in May 1987 as Licensing Topical

Report NEDO-31400, titled "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor" (Reference 1). The report was approved by the NRC in a Safety Evaluation Report sent to the Owners Group on May 15, 1991 (Reference 2). Elimination of these trip functions provides the following safety-related enhancements:

1. Reduction in Scram Frequency

   Elimination of the trip functions will reduce the exposure to spurious trips. This is especially true when calibration of reactor scram instrumentation, including the main steam line high radiation monitors, is in progress since this activity involves inserting a half scram signal. As noted in NEDO-31400, eight scrams have been attributed to the MSLRM trip system since 1980. Failure of one of the MSLRMs was a contributing factor to an automatic reactor scram of the FitzPatrick plant on May 25, 1993. This event is described in LER 93-013. Removing the trips would represent a reduction in transient initiating events. As described in NEDO-31400, an evaluation using MSLRM initiated scram data for BWR plants, results in a 0.3% reduction in the generic core damage frequency probability.

2. Maintain Availability of Condenser Heat Sink

   Elimination of this MSIV closure function will maintain availability of the main condenser as a heat sink to facilitate scram recovery.

3. Elimination of the Potential for Trips Due to Hydrogen Water Chemistry

   Hydrogen Water Chemistry injection rates influence nitrogen-16 levels in the main steam lines, introducing the possibility that system transients could momentarily increase nitrogen-16 levels to the point that a high radiation trip could occur.

4. Increased Operator Control Over Radioactive Releases

   Elimination of this MSIV closure function allows the Steam Jet Air Ejectors to remain operable for a longer period of time. This permits the continued use of the Offgas Treatment System to process radioactivity during transients that may occur.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

The bases for approval of the proposed changes is the NRC's Safety Evaluation Report approving the BWR Owners Group License Topical Report NEDO-31400 (Reference 2). The SER concluded that the removal of the MSLRM trips that automatically shutdown the reactor and close the MSIVs is acceptable provided the plant meets three conditions. These conditions are discussed later in this section.

As stated in NEDO-31400, and indicated in the FitzPatrick Final Safety Analysis Report, the automatic reactor shutdown on the MSLRM trip is not given credit in the analysis of any design basis event for BWRs. The FSAR assumes that the MSIVs, on the MSLRM trip, close only in a control rod drop accident (CRDA). However, the Standard Review Plan (SRP) 15.4.9, Rev. 2, July 1981, recommends an assumption that the radioactive contents (noble and iodine) of the coolant resulting from the event are transferred to the condenser and turbine before the MSIVs close.

SRP 15.4.9 states that the plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated control rod drop accident if the calculated whole-body and thyroid doses at the exclusion area boundaries are well within the exposure guideline values in 10 CFR 100. "Well within" is defined in SRP 15.4.9 as 25% of the 10 CFR 100 exposure guideline values. This translates to guideline values of 75 Rem for thyroid doses and 6 Rem for whole-body doses.

NEDO-31400 confirmed that the radiological release consequence without the automatic MSIV trip is within the NRC's acceptance criteria as stated in SRP 15.4.9. NEDO-31400 analyzes two scenarios for the CRDA as follows:

Scenario 1:

This is the FSAR bounding scenario that involves automatic MSIV closure. This scenario assumes that the fission product activity is airborne in the turbine and condenser following MSIV closure and leaks directly from the condenser to the atmosphere.

Scenario 2:

This scenario assumes that no automatic MSIV closure occurred and the fission products are transported to an augmented offgas system. The release of the activity to the environment would be from the normal offgas release point after holdup in the treatment system.

Calculations of post-accident doses for the site boundary were performed for each of these scenarios to compare radiological consequences with the exposure guidance of SRP 15.4.9. The calculations, using the conservative assumptions of SRP 15.4.9, yield for both scenarios site boundary doses which are a small fraction of 10 CFR 100 and SRP 15.4.9 guidelines. Since the plant specific doses are a function of the plants dispersion coefficient (X/Q) and offgas treatment system holdup times, NEDO-31400 provides curves for computing a plant specific value. Fitzpatrick utilizes an augmented offgas treatment system. The specific values of X/Q and offgas holdup times for the FitzPatrick Nuclear Power Plant are shown on Table 1.

Plant Specific Analysis

Site boundary doses were calculated for the James A. FitzPatrick Nuclear Power Plant (Reference 5) for the scenarios with and without MSIV closure (Scenarios 1 and 2, respectively). The site boundary doses are presented in Table 2. Two methods were used to calculate the plant specific site boundary doses. One method used in-house analysis capabilities, the other method used the curves in NEDO-31400. There is good agreement between the in-house calculations and the NEDO-31400 curves. Plant specific values were used for the power level of the failed fuel rods, X/Q, and offgas holdup times. The assumptions are bounded by NEDO-31400. The in-house calculation assumed a more conservative depletion factor for halogens in the turbine and condenser by partitioning and plateout than the NEDO-31400 value (50% instead of 90%). The doses presented in Table 2 have been corrected for the 90% depletion assumed in NEDO-31400. Under either scenario, the doses at the site boundary are well within the acceptance criteria of SRP 15.4.9 guidelines. The doses presented in Table 2 for the design basis CRDA were obtained from the FitzPatrick FSAR, Section 14.6.1.2.

The plant specific doses calculations were performed using power uprate initial conditions. These conditions are more conservative for the current licensed power level, and the calculations will not need to be revised to accommodate anticipated NRC approval of power uprate. Power uprate was requested in Technical Specification amendment request, JPTS-91-025 (Reference 6).

NRC Conditions

The NRC in a May 15, 1991 Safety Evaluation Report concluded that removal of the MSLRM trips that automatically shutdown the reactor and close the MSIVs is acceptable provided the licensee references NEDO-31400 in support of their licensing applications and meets the following conditions:

1. "The applicant demonstrates that the assumptions with regard to input values (including power per assembly, X/Q, and decay times) that are made in the generic analysis bound those for the plant."

   NYPA Position

   The assumptions made in the generic analysis (NEDO-31400) bound those used in the plant specific analysis (Reference 5). Plant specific values used in the FitzPatrick analysis are shown on Table 1.

2. "The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases."

NYPA Position

Reasonable assurance is provided in the plant response to increased radiation levels as detected by the offgas monitor (located between the Steam Jet Air Ejector (SJAE) and the Offgas treatment system). Abnormal Operating Procedure, AOP-03, "High Activity in Reactor Coolant or Offgas," currently controls the plant response. The offgas monitor is a more sensitive monitor than the MSLRM because the N-16 source, dominating the radiation source surveyed by the MSLRM, has decayed by the time the offgas monitor can be affected by any increased levels of activity. As required by the FitzPatrick Technical Specifications (Appendix B, Specification 3.5), a level of 500 millicuries/second, after a 15 minute delay, will close the SJAE isolation valve, and will prompt a power reduction to return to acceptable levels within 72 hours or a shutdown within an additional 12 hours if the limit cannot be met. This graded response ensures that actions are taken to limit occupational doses and environmental releases. Prior to modification of the MSLRM trip, plant procedures will be reviewed, and revised as appropriate.

The FitzPatrick Operating License permits bypassing of the Offgas Treatment System during plant startup. NEDO-31400 states that this condition is acceptable provided the offgas radiation monitors are being utilized to automatically isolate the offgas process line. The SJAE isolation feature, previously described, conforms with this NEDO-31400 criteria, and precludes a direct release to the environment. In the event the offgas system is isolated, the offgas dose is equivalent to the FSAR design basis scenario (with MSIV isolation) since in this case the activity is assumed to be transferred to the main condenser, followed by a ground level release. The NRC conditions stipulated in their SER for the offgas radiation monitor will be implemented as describe below for condition 3.

3. "The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the nominal nitrogen-16 background dose rate at the monitor location, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints."

NYPA Position

Concurrent with the modification of the MSLRM trip, the alarm setpoints on the MSLRM and offgas radiation monitor will be adjusted to less than or equal to 1.5 times the normal full power N-16 background dose rate (accounting for the increased N-16 carryover due to hydrogen water chemistry). Prior to modification of the MSLRM trip, the plant procedures will be revised to require prompt sampling of the reactor coolant to determine the need for corrective actions, if the MSLRM or offgas radiation monitors, or both, exceed their alarm setpoints.

Conclusion

There are no adverse safety implications associated with removal of the MSLRM scram and MSIV closure function since the offsite radiation exposure levels without the trips are comparable to those associated with the trip function, and are well within 10 CFR 100 and SRP 15.4.9 guidelines. The analysis has been performed in accordance with an NRC approved Licensing Topical Report NEDO-31400 and conforms with conditions imposed by the NRC in a Safety Evaluation Report. The MSLRM isolation function is retained for the main steam drain valves, the recirculation loop sample valves, and the mechanical vacuum pumps and associated isolation valves. The Technical Specifications for the offgas radiation monitor, and associated trip function, are not affected by this application.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick 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.

   None of the FitzPatrick design basis events takes credit for reactor scram initiated from the MSLRM. Therefore, elimination of the scram trip signal will not increase the probability or consequences of accidents previously evaluated.

   The Control Rod Drop Accident (CRDA) is the only design basis event which assumes that the reactor vessel isolation signal comes from the MSLRM. The isolation trip will not prevent the CRDA from occurring, therefore its elimination will not increase the probability of the accident.

   The MSLRM isolation of the MSIVs was intended to mitigate the consequences of a CRDA. The fission products transported to the main condenser before MSIV closure, results in a ground level release due to condenser leakage. However, NEDO-31400, and the plant specific analysis using the NEDO-31400 assumptions and methodology, demonstrates that the isolation is actually of little benefit in this regards. Without MSIV isolation, the steam jet air ejector remains operational, and the fission products are processed through the augmented offgas treatment system. The holdup time, charcoal adsorption, and elevated release, provided by the offgas treatment system, limits the offsite exposure levels. The analysis shows the offsite thyroid doses for the CRDA reduced to zero without MSIV closure. There is a small increase in the whole body doses without MSIV closure; however, the conservatively calculated values are a small fraction of 10 CFR 100 and SRP 15.4.9 guidelines.

Therefore, the elimination of the MSLRM isolation trip will not increase the consequences of an accident previously evaluated.

2. create the possibility of a new or different kind of accident from those previously evaluated.

The MSLRM scram and MSIV isolation, were originally intended to mitigate, not prevent an accident scenario. Other than the circuitry modifications required to accomplish the removal of the subject trips, no changes to the physical plant or to the manner in which the plant is operated are introduced by the requested change. The change does not affect the remaining scram or vessel isolation functions. Therefore, no new or different kind of accident is created.

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

The Licensing Topical Report NEDO-31400, as approved by the NRC, provides the results of a reliability assessment of the elimination of the MSLRM scram function on reactivity control failure frequency and core damage frequency. The results of the analysis indicate a negligible increase, on a generic basis, in reactivity control failure frequency with the deletion of the MSLRM scram function (1.4 E-9 events/year). However, this increase in reactivity control failure frequency is offset by the reduction in the transient initiating events (inadvertent scrams). This reduction in transient initiating events represents a 0.3% reduction in the generic core damage frequency (Reference 1).

Safe operation of the plant is further enhanced by elimination of the unnecessary scram and subsequent isolation of the reactor vessel. With implementation of these changes, the primary heat sink (main condenser) remains available, a large transient on the vessel and safety-related actuations are avoided, and the Offgas Treatment System remains available to control the potential release pathway.

The existing MSLRM and offgas radiation monitoring instrumentation will remain in service to provide information and alarms to plant operators. In the event either or both of these monitors alarm, the reactor coolant will be promptly sampled to determine activity levels and the need for additional corrective actions. The offgas treatment system isolation trip function on high radiation remains unaffected by this change. The MSLRM isolation functions, other than for the MSIV's, also remain unaffected by this change.

For these reasons, the proposed changes will enhance the margin of safety.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

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

The trip will be removed following NRC approval of the proposed Technical Specification change.

VI. CONCLUSION

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

1. will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

2. will not increase the possibility of an accident or malfunction of a type different from any previously evaluated in the Safety Analysis Report; and

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

The changes therefore involve no signification hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. Licensing Topical Report NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam line Radiation Monitor," May 1987.

2. NRC Safety Evaluation report, letter from Ashok C. Thadani, NRC to George J. Beck, BWROG, "Acceptance for Referencing of Licensing topical Report NEDO-31400," May 15, 1991.

3. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report.

4. USNRC Standard Review Plan 15.4.9, Rev. 2, July 1981, "Radiological Consequences of Control Rod Drop Accident (BWR).

5. New York Power Authority Calculation: JAF-CALC-RAD-00013, "Radiological Justification for Modification of the Main Steam Line Radiation Monitor Trip Functions," March 13, 1992.

6. NYPA letter, R. E. Beedle to NRC, JPN-92-028, Proposed Changes to the Technical Specifications Regarding Power Uprate (JPTS-91-025), dated June 5, 1992.

TABLE 1

PLANT SPECIFIC VALUES
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

Offgas System Holdup Times

Kryptons:	4.6 hours
Xenons:	106.7 hours
Halogens:	100% removal (34,250 lbs. of charcoal in beds)

Site Boundary Dispersion Coefficients*

	Time Interval (hrs)	Conc. X/Q (sec/m^3)	Gamma X/Q (sec/m^3)
Stack Release	0 - 2**	5.24E-5	4.75E-5
Condenser Release	0 - 2	1.81E-4	1.31E-4

* Concentration X/Q is for thyroid exposure due to inhalation. Gamma X/Q is for whole body external gamma exposure from a finite plume.

** Fumigation conditions.

Other Plant Specific Values / Assumptions

Reactor Power	2585.7 MWt
Power Level of Failed Fuel Rods	0.106 MWt/rod
Radial Peaking Factor	1.5
No. of Failed Fuel Rods	850
Main Condenser Leakage	1%/day (Scenario 1 only)
Offgas System Flowrate	60 cfm
Fission Products Released From Failed Rods	10% halogens 10% noble gases (except Kr-85), 30% Kr-85

TABLE 2

SITE BOUNDARY RADIATION EXPOSURES
CONTROL ROD DROP ACCIDENT
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

MODEL	THYROID (Rem)	WHOLE BODY (Rem)
SRP 15.4.9 Limit	75	6
Design Basis CRDR (FSAR)	3.94	0.242
NEDO-31400, Scenario 1* (MSIV Isolation)	0.323 (0.32)	0.013** (0.015)
NEDO-31400, Scenario 2* (No MSIV Isolation)	0	1.457 (2.2)

* First value is from reference 5, adjusted for 90% depletion of halogens in the turbine and condenser, as assumed in NEDO-31400. Value in parenthesis computed from curves in NEDO-31400.

** 0.0123 rem from noble gases, and 0.0006 rem from halogens.

ATTACHMENT III to JPN-93-050

MARKUP OF TECHNICAL SPECIFICATION PAGES FOR
PROPOSED TECHNICAL SPECIFICATION CHANGES
ELIMINATION OF THE MAIN STEAM ISOLATION VALVE CLOSURE FUNCTION
AND SCRAM FUNCTION OF THE MAIN STEAM LINE RADIATION MONITOR

JPTS-90-008

New York Power Authority

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

Attachment III to JPN-93-050

INSERTS FOR MARKUP TECHNICAL SPECIFICATION PAGES

INSERT A

High radiation monitors in the area of the main steam lines have been provided to detect gross fuel failure as in the control rod drop accident. A trip setting of 3 times normal full-power background is established to close the main steam line drain valves, the recirculation loop sample valves, and the mechanical vacuum pump isolation valves, and trip the pumps, to limit fission product release. For changes in the Hydrogen Water Chemistry hydrogen injection rate, the trip setpoint may be adjusted based on a calculated value of the expected radiation level. Hydrogen addition will result in an increase in the N-16 carryover in the main steam.

INSERT B

Isolate the main steam line drain valves, the recirculation loop sample valves, and the mechanical vacuum pump, within eight hours.

INSERT C

Perform a calibration once per operating cycle using a radiation source. Perform an instrument channel alignment once every 3 months using the built-in current source.

subchannel. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, main steam isolation valve (MSIV) closure, and generator load rejection, turbine stop valve closure are discussed in Sections 2.1 and 2.2.

Instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the Core and Containment Cooling Systems (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

High radiation levels in the main steam line tunnel, above normal levels due to the nitrogen and oxygen radioactivity, are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector offgas monitors which cause an isolation of the main condenser offgas line. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by a factor of approximately 5 at the maximum hydrogen addition rate as indicated in note 16, Table 3.1-1. The scram setpoint will be reset to three times the projected background radiation level prior to performance of the test. The setpoint will be restored to normal following completion of the hydrogen addition test.

A Reactor Mode Switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference paragraph 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (high flux in startup or refuel) System provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The IRM System provides protection against short reactor periods in these ranges.

The Control Rod Drive Scram System is designed so that all of the water which

Move to page 34

3.1 BASES (cont'd)

Relocate text from page 33

is discharged from the reactor by a scram can be accommodated in the discharge piping. Each scram discharge instrument volume accommodates in excess of 34 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches 34.5 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR).

The IRM high flux and APRM $\leq$ 15% power scrams provide adequate coverage in the startup and intermediate range. Thus, the IRM and APRM systems are required to be operable in the refuel and startup/hot standby modes. The APRM $\leq$ 120% power and flow referenced scrams provide required protection in the power range (reference FSAR Section 7.5.7). The power range is covered only by the APRMs. Thus, the IRM system is not required in the run mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below 217 psig turbine first stage pressure (30 percent of rated), the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting [1]	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)	Startup	Run		
2	APRM Downscale	$\geq$ 2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	$\leq$ 1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	$\leq$ 2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	$\geq$ 177 in. above TAF	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	$\leq$ 34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
2	Main Steam Line High Radiation	$\leq$ 3x normal full power background (16)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	$\leq$ 10% valve closure			X(5)	8 Instrument Channels	A

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TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1- (cont'd)

C. High Flux IRM.

D. Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted.

E. APRM 15% Power Trip.

7. Not required to be operable when primary containment integrity is not required.
8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
9. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.
10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.
11. See Section 2.1.A.1.
12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit specified in the Core Operating Limits Report.
13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained as specified in the Core Operating Limits Report.
14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.
15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.
16. *During the proposed Hydrogen Addition Test, the background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, within 24 hours prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to $\leq$ three times the anticipated radiation levels. Upon completion of the Hydrogen Addition Test, the setpoint will be readjusted to its prior setting within 24 hours.

* This specification is in effect only during Operating Cycle 10.

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Table 4.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TEST
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/week.
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/month. (1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Once/month.
Turbine First Stage Pressure Permissive	B	Trip Channel and Alarm (4)	Once/month. (1)(8)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/month. (1)

NOTES FOR TABLE 4.1-1

1. Initially once every month until acceptable failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of JAFNPP.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

   If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the instrument channels.

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	Every refueling outage
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	B	Standard Pressure Source	Note (6)
High Drywell Pressure	B	Standard Pressure Source	Note (6)
Reactor Low Water Level	B	Standard Pressure Source	Note (6)
High Water Level in Scram Discharge Instrument Volume	A	Water Column, Note (5)	Once/operating cycle, Note (5)
High Water Level in Scram Discharge Instrument Volume	B	Standard Pressure Source	Every 3 months
Main Steam Line Isolation Valve Closure	A	Note (4)	Note (4)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months
Turbine First Stage Pressure Permissive	B	Standard Pressure Source	Note (6)

TABLE 4.1-2 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	Once/operating cycle
Turbine Stop Valve Closure	A	Note (4)	Note (4)

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.
2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.
3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be performed each refueling outage.
4. Actuation of these switches by normal means will be performed during the refueling outages.
5. Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.
6. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

Insert: "Deleted"

Insert A

3.2 BASES (cont'd)

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.1.2 FSAR. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by approximately a factor of 5 at the peak hydrogen concentration as indicated in note 16, Table 3.1-1. With the hydrogen addition, the fission product release would still be well within the 10 CFR 100 guidelines in the event of a control rod drop accident.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drops below 825 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of approximately 300 percent of design flow for this high flow or 40°F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip settings of approximately 300 percent for high flow or 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the Safety Limit. The trip

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TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (2)
2 (6)	Reactor Low Water Level	$\geq$ 177 in. above TAF	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	$\leq$ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low-Low Water Level	$\geq$ 18 in. above TAF	4 Inst. Channels	A
2 (6)	High Drywell Pressure	$\leq$ 2.7 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	< 3 x Normal Rated Full Power Background (9)	4 Inst. Channels	B Insert "E"
2	Low Pressure Main Steam Line	$\geq$ 825 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	$\leq$ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	$\leq$ 40°F above max ambient	4 Inst. Channels	B
4	Reactor Cleanup System Equipment Area High Temperature	$\leq$ 40°F above max ambient	8 Inst. Channels	C
2	Low Condenser Vacuum Closes MSIV's	$\geq$ 8" Hg. Vac (7)(8)	4 Inst. Channels	B

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TABLE 3.2-1 (Cont´d)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.
2. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate shutdown cooling.

   E. Insert B

3. Deleted
4. Deleted
5. Two required for each steam line.
6. These signals also start SBGTS and initiate secondary containment isolation.
7. Only required in run mode (interlocked with Mode Switch).
8. Bypassed when mode switch is not in run mode and turbine stop valves are closed.
9. The trip level setpoint will be maintained at $\leq$ 3 times normal rated full power background. See note 16 to Table 3.1-1 for re-setting trip level setpoint just prior to and following the Hydrogen Addition Test.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (8)	Instrument Functional Test	Calibration Frequency	Instrument Check (4)
1)	Reactor High Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2)	Reactor Low-Low-Low Water Level	(1)(5)	(15)	Once/day
3)	Main Steam High Temp.	(1)(5)	(15)	Once/day
4)	Main Steam High Flow	(1)(5)	(15)	Once/day
5)	Main Steam Low Pressure	(1)(5)	(15)	Once/day
6)	Reactor Water Cleanup High Temp.	(1)	Once/3 months	None
7)	Condenser Low Vacuum	(1)(5)	(15)	Once/day
Insert 8)	Main Steam Line High Radiation	(1)(5)	(11)	Once/day

	Logic System Functional Test (7) (9)	Frequency
1)	Main Steam Line Isolation Valves Main Steam Line Drain Valves Reactor Water Sample Valves	Once/6 months
2)	RHR - Isolation Valve Control Shutdown Cooling Valves	Once/6 months
3)	Reactor Water Cleanup Isolation	Once/6 months
4)	Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves	Once/6 months
5)	Standby Gas Treatment System Reactor Building Isolation	Once/6 months

NOTE: See notes following Table 4.2-5.

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NOTES FOR TABLES 4.2-1 THROUGH 4.2-5

1. Initially once every month until acceptance failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instruments operate in a environment similar to that of JAFNPP.

2. Functional tests are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed within seven (7) days prior to each startup.

3. Calibrations are not required when these instruments are not required to be operable or are tripped. Calibration tests shall be performed within seven (7) days prior to each startup or prior to a pre-planned shutdown.

4. Instrument checks are not required when these instruments are not required to be operable or are tripped.

5. This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

6. These instrument channels will be calibrated using simulated electrical signals once every three months.

7. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

8. Reactor low water level, high drywell pressure ~~and high radiation main steam line tunnel~~ are not included on Table 4.2-1 since they are tested on Table 4.1-2.

9. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

10. At least one (1) Main Stack Dilution Fan is required to be in operation in order to isokinetically sample the Main Stack.

11. Uses same instrumentation as Main Steam Line High Radiation. See Table 4.1-2.

12. (Deleted)

13. Calibration and instrument check surveillance for SRM and IRM Instruments are as specified in Tables 4.1-1, 4.1-2, 4.2-3.

14. Functional test is performed once each operating cycle.

15. Sensor calibration once per operating cycle. Master/slave trip unit calibration once per 6 months.

Insert C

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

(f) Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Tables 4.1-1 and 4.1-2 respectively). These requirements are performed as part of main steam high radiation monitor surveillances.

Insert: Table 4.2-1

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.