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	≥ 12.5 in. indicated level (≥ 177 in. above the top of active fuel)	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
2	Main Steam Line High Radiation	≤ 3x normal full power background (16)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ 10% valve closure	X(3)(5)	X(3)(5)	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)

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 placed in the Run position.

16. During the proposed Hydrogen Addition Test, the normal background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to ≤ three times the increased radiation levels. The test will be conducted at power levels > 80% of normal rated power. During controlled power reduction, the setpoint will be readjusted prior to going below 20% rated power. If due to a recirculation pump trip or other unanticipated power reduction event, the reactor drops below 20% rated power without the setpoint change, control rod withdrawal will be prohibited until the necessary trip setpoint adjustment is made.

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum Number 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	≥ 12.5 in Indicated Level (≥ 177 in. above the top of active fuel)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low Water Level	≥ -38 in. indicated level (≥ 126.5 in. above the top of active fuel)	4 Inst. Channels	A
2(6)	High Drywell Pressure	≤ 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
2	Low Pressure Main Steam Line	≥ 825 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	≤ 40°F above max ambient	4 Inst. Channels	B
3	Reactor Cleanup System Equipment Area High Temperature	≤ 40°F above max ambient	6 Inst. Channels	C
2	Low Condenser Vacuum Closes MSIV's	≥ 8" Hg. Vac (8)	4 Inst. Channels	B

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.

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 reactor pressure is less than 1005 psig 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 the Hydrogen Addition Test, and re-setting of the Main Steam Line Radiation Monitor for power levels below 20%.

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 (pressure switches) for the drywell are 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 that due to the normal 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

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 drop 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 $\leq$ 300 percent of design flow for this high flow of 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 setting of $\leq$ 300 percent for high flow and 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high flow 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 de-

ATTACHMENT II

Safety Evaluation of Proposed Change to
Technical Specifications - Main Steam Line Radiation
Monitor Trip Level Setpoint Related to Hydrogen Addition Test

New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333

Section I Description of the Change

The proposed change to the Technical Specifications is shown in Attachment I to the Amendment Application. This change occurs in Table 3.1-1 pages 41a and 43a, Table 3.2-1 pages 64 and 65 and in the Bases on pages 33 and 57. The change on the above mentioned pages alters the main steam line radiation monitor trip level setting to take into account the higher background radiation level due to hydrogen addition to the primary coolant.

Section II Purpose of the Change

The purpose of the change is to avoid spurious trips of the main steam line radiation monitor during the hydrogen addition test. During this test, the addition of hydrogen reduces the concentration of oxygen in the coolant water and increases the N-16 carryover in the steam. This results in a higher background radiation level seen by the main steam line radiation monitor, which would be above the existing trip setpoint.

Section III Impact of the Change

The main steam line radition monitors have only a single design basis which is to initiate a reactor scram and isolate the main steam lines upon detecting high radiation, due to gross fission product release during a control rod drop accident (CRDA). This is discussed in FSAR Sections 7.2.3.6, 7.3.4.8 and 7.12. The results of a CRDA are more severe at power levels $<$ 10% as stated in FSAR Section 14.6.1.2 and the hydrogen addition test will be conducted at power levels $>$ 80% of rated power. If, due to a recirculation pump trip or any other unanticipated power reduction event, the reactor power decreases below 20% of rated power, control rod withdrawal will be prohibited until the necessary re-adjustment is made to the trip setpoint.

The licensing basis for the CRDA states that the maximum control rod worth is established by assuming the worst single inadvertent operator error (Reference d). Assuming this operator error, References c and d establish the maximum control rod worth above 20% of rated power. Parametric studies utilizing the conservative GE excursion model (Reference b) indicate that the maximum peak fuel enthalpy for a dropped control rod of maximum worth is less than 120 calories per gram. Consequently, the conservatively calculated peak fuel enthalpy for a CRDA above 20% of rated power will have significant margin to the fuel cladding failure threshold of 170 calories per gram.

The proposed change to the Technical Specifications will not alter the conclusions reached in the FSAR and SER accident analyses.

The Authority considers that this proposed change can be classified as not likely to involve a significant hazards consideration since this proposed change [as per 10 CFR 50.92 (c)(1), (2) and (3)]:

1. does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the proposed test would be conducted at power levels $>$ 80% of rated power. Above 20% of rated power, there is a significant margin between the calculated peak fuel enthalpy and the fuel cladding failure threshold enthalpy. For power levels below 10% of rated power for which the CRDA results become more severe, the trip setpoint will be readjusted to the original setting, as discussed above;

2. does not create the possibility of a new or different kind of accident from any accident previously evaluated, because the only function of these monitors is to detect gross fission product release in the event of a CRDA. Below 20% of rated power, the monitors would be at their original setting. Above 20% of rated power, there will be a significant margin to the fuel cladding failure threshold; and

3. does not involve a significant reduction in margin of safety because the monitor setpoints will only be changed above 20% of rated power, and a significant margin of safety will still exist.

For these reasons the proposed change is similar to the example (vi) included in Federal Register, Vol. 48 No. 67 dated April 6, 1983, Page 14870. This Commission example is "(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan". The proposed change is similar to this example because the results of the change are within all acceptable criteria.

Section IV <u>Implementation of the Change</u>

The change as proposed will not impact:

1. Radiation/ALARA Considerations

   Normal radiation and ALARA practices and procedures will be in effect during the course of the test. Appropriate approved access controls will be implemented for areas subject to the higher radiation levels that result from the test. Dose rate surveys will be conducted and radiation levels will be monitored in order to comply with ALARA requirements.

2. Fire Protection

Hydrogen monitors will be installed to detect hydrogen leaks. Sufficient oxygen will be injected upstream of the recombiner unit to assure efficient hydrogen recombination. In the unlikely event of any difficulty, the test could be terminated immediately.

3. Environment

There will be no significant impact on the environment.

Section V Conclusion

The incorporation of this change; a) will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as evaluated previously in the Safety Analysis Report; b) will not increase the possibility of an accident or malfunction of a type other than that evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the Bases for any Technical Specification; d) does not constitute an unreviewed safety question and; e) involves no significant hazards consideration, as defined in 10 CFR 50.92.

Section VI References

a) JAF FSAR Sections 7.2.3.6, 7.3.4.8, 7.12 and 14.6.1.2

b) R.C. Stirn, et al. Rod Drop Analysis for Large Boiling Water Reactors, General Electric Company, March, 1972 (NEDO-10527)

c) R.C. Stirn, et al. Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 1 Multiple Enrichment Cores with Axial Gadolinium, General Electric Company, July, 1972 (NEDO-10527, Supplement 1)

d) R.C. Stirn, et al. Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores, General Electric Company, January, 1973 (NEDO-10527, Supplement 2)