3.1.1.5 Pressurizer

Whenever the reactor is at hot shutdown or critical, the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hours or have the RHR system in operation within an additional 6 hours.

3.1.1.6 Reactor Coolant System Vents

a. When the reactor is at hot shutdown or critical, at least one reactor coolant system vent path consisting of two valves in series shall be operable and closed* at each of the following locations:

1. Reactor Vessel head
2. Pressurizer steam space

*The PORV block valve is not required to be closed but must be operable if the PORV is capable of being opened.

b. With one or more vents at the above reactor coolant system vent path locations inoperable, startup may commence and/or power operation may continue provided at least one vent path is operable and the inoperable vent paths are maintained closed with motive power removed from the valve actuator of all the valves in the inoperable vent paths. If the requirements of 3.1.1.6a are not met within 30 days, be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

c. With all of the above reactor coolant system vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 72 hours or be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

Bases

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal

operations and anticipated transients. Heat transfer analyses[1] show that reactor heat equivalent to 130 MWT (8.5%) can be removed by natural circulation alone. Therefore operation with one operating reactor coolant loop while below 130 MWT provides adequate margin.

The specification permits an orderly reduction in power if a reactor coolant pump is lost during operation between 130 MWT and 50% of rated power.[2] Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than one which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase.

When the reactor coolant system average temperature is above 350°F, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require one loop be in operation and the other loop be capable of removing heat via natural circulation.

When the reactor coolant system average temperature is between 200°F and 350°F or while in cold shutdown, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity

changes in the reactor. Mixing of the reactor coolant will be sufficient to prevent a sudden increase in reactivity if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant. When the boron concentration of the reactor coolant system is to be increased, the process must be uniform to prevent sudden reactivity increases in the reactor during subsequent startup of the reactor coolant pumps. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump is running while the change is taking place. Emergency boration without a reactor coolant pump in operation is not prohibited by this specification.

Prohibiting reactor coolant pump starts without a large void in the pressurizer or without a limited RCS temperature differential will prevent RCS overpressurization due to expansion of cooler RCS water as it enters a warmer steam generator. A 38% level in the pressurizer will accommodate the swell resulting from a reactor coolant pump start with a RCS temperature of 140°F and steam generator secondary side temperature of 340°F, or the maximum temperature which usually exists prior to cooling the reactor with the RHR system.

Temperature requirements for the steam generator correspond with measured NDT for the shell and allowable thermal stresses in the tube sheet.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve set point. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provides adequate defense against overpressurization.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. The requirement that 100 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown and during cooldown.(3)

Reactor Coolant System Vents

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head and one from the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

References

(1) FSAR Section 14.1.6
(2) FSAR Section 7.2.3
(3) Letter from L.D. White, Jr. to D. L. Ziemann, USNRC, dated October 17, 1979

3.5 **Instrumentation Systems**

Applicability:

Applies to plant instrumentation systems.

Objective:

To delineate the conditions of the plant instrumentation and safety circuits and to limit the release or radioactive materials.

Specification:

3.5.1 Operational Safety Instrumentation

3.5.1.1 The number of Minimum Operable Channels for instrumentation shown on Tables 3.5-1 through 3.5-3 shall be OPERABLE for plant operation at rated power.

3.5.1.2 In the event the number of channels of a particular subsystem in service falls below the limit given in the columns entitled Minimum Operable Channels, operation shall be limited according to the requirement shown in the last column of Tables 3.5-1 through 3.5-3.

3.5.2 Accident Monitoring Instrumentation

3.5.2.1 The accident monitoring instrumentation channels shown in Table 3.5-4 shall be operable whenever the reactor is at hot shutdown or is critical.

3.5.2.2 While critical, with the number of operable accident monitoring instrumentation channels less than the Total Required Number of Channels shown in Table 3.5-4, either restore the inoperable channel(s) to operable status within 7 days, or be in at least hot shutdown within the next 12 hours.

3.5.2.3 While critical, with the number of operable accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.5-4, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

3.5.2.4 The radiation accident monitoring instrumentation channels shown in Table 3.5-7 shall be operable whenever the reactor is at hot shutdown or is critical. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.5-7. Startup may commence or continue consistent with the action statement.

3.5.3 Engineered Safety Feature Actuation Instrumentation

3.5.3.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Tables 3.5-2 and 3.5-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5-5.

3.5.3.2 With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5-5, declare the channel inoperable and apply the applicable ACTION requirement of Tables 3.5-2 and 3.5-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

3.5.3.3 With an instrumentation channel inoperable, take the action shown in Tables 3.5-2 and 3.5-3.

3.5.4 Radioactive Effluent Monitoring Instrumentation

3.5.4.1 The radioactive effluent monitoring instrumentation shown in Table 3.5-6 shall be operable at all times with alarm and/or trip setpoints set to ensure that the limits of Specifications 3.9.1.1 and 3.9.2.1 are not exceeded. Alarm and/or trip setpoints shall be established in accordance with calculational methods set forth in the Offsite Dose Calculation Manual.

3.5.4.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

(i) the setpoint shall be immediately corrected without declaring the channel inoperable; <u>or</u>

(ii) immediately suspend the release of effluents monitored by the affected channel; <u>or</u>

(iii) declare the channel inoperable.

3.5.4.3 If the number of channels which are operable is found to be less than required, take the action shown in Table 3.5-6.

3.5.5 <u>Control Room HVAC Detection Systems</u>*

3.5.5.1 During all modes of plant operation, detection systems for chlorine gas, ammonia gas and radioactivity in the control room HVAC intake shall be operable with setpoints to isolate air intake adjusted as follows:

chlorine, $\leq$ 5 ppm

ammonia, $\leq$ 35 mg/m^3

radioactivity, particulate $\leq 1 \times 10^{-8}$ uc/cc

iodine $\leq 9 \times 10^{-9}$ uc/cc

noble gas $\leq 1 \times 10^{-5}$ uc/cc

3.5.5.2 With one of the detection systems inoperable, within 1 hour isolate the control room HVAC air intake. Maintain the air intake isolated except for short periods, not to exceed 1 hour a day, when fresh air makeup is allowed to improve the working environment in the control room.

* Effective January 1, 1985

Basis

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendation".

The radioactive liquid effluent instrumentation is provided to monitor and/or control, as applicable, the releases of radioactive materials in liquid effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR Part 50.

Reference

FSAR - Section 7.2.1

TABLE 3.5-4

Accident Monitoring Instrumentation

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS (7)	MINIMUM CHANNELS OPERABLE (7)
1. Pressurizer Water Level (1)	2	1
2. Auxiliary Feedwater Flow Rate (2)(3)	2/steam generator	1/steam generator
3. Steam Generator Water Level - Wide Range (3)	1/steam generator	1/steam generator
4. Reactor Coolant System Subcooling Margin Monitor (4)	2	1
5. Pressurizer PORV Position Indicator (5)	2/Valve	1/Valve
6. PORV Block Valve Position Indicator (1)	1/Valve	0/Valve
7. Pressurizer Safety Valve Position Indicator (5)	2/Valve	1/Valve
8. Containment Pressure (8)	2	1
9. Containment Water Level (Narrow Range, Sump A)	1(6)	1(6)
10. Containment Water Level (Wide Range, Sump B)	2	1
11. Core-Exit Thermocouples	4/core quadrant	2/core quadrant

Notes

(1) Emergency power for pressurizer equipment, NUREG-0737, item II.G.1.

(2) Auxiliary feedwater system flow indication, NUREG-0737, item II.E.1.2.

(3) Only 2 out of the 3 indications (two steam generator auxiliary feedwater flow and one wide-range steam generator level) are required to be operable, NUREG-0737, item II.E.1.2.

(4) Instrumentation for detection of inadequate core cooling, NUREG-0737, item item II.F.2.1.

(5) Direct indication of relief and safety valve position, NUREG-0737, item II.D.3. Two channels include a primary detector and RTD as the backup detector.

(6) Operation may continue with less than the minimum channels operable provided that the requirements of Technical Specification 3.1.5.3 are met.

(7) See Specification 3.5.2 for required action.

(8) Containment pressure monitor, NUREG-0737, item II.F.1.4.

[illegible]

TABLE 3.5-7

Radiation Accident Monitoring Instrumentation

Instrument	Minimum Channels Operable	Action
1. Containment Area (R-29 and R-30)	2	1
2. Noble Gas Effluent Monitors		
i. Plant Vent (R-14A)	1	1
ii. A Main Steam Line (R-31)	1	1
iii. B Main Steam Line (R-32)	1	1
iv. Containment Purge (R-12A)	1	1
v. Air Ejector (R-15A)	1	1

Action Statements

Action 1 - With the number of operable channels less than required by the Minimum Channels Operable requirements, either restore the inoperable channel(s) to operable status within 7 days of the event, or prepare and submit a Special Report to the Commission within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

3.6.3 Containment Isolation Valves

3.6.3.1 With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve operable in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to operable status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Isolation valves are inoperable from a leakage standpoint if the leakage is greater than that allowed by 10 CFR 50 Appendix J.

3.6.4 Combustible Gas Control

3.6.4.1 When the reactor is critical, at least two independent containment hydrogen monitors shall be operable. One of the monitors may be the Post Accident Sampling System.

3.6.4.2 With only one hydrogen monitor operable, restore a second monitor to operable status within 30 days or be in at least hot shutdown within the next 6 hours.

3.6.4.3 With no hydrogen monitors operable, restore at least one monitor to operable status within 72 hours or be in at least hot shutdown within the next 6 hours.

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
25.	Containment Pressure	S	R	M	Narrow range containment pressure (-3.0, +3 psig) excluded
26.	Steam Generator Pressure	S	R	M	
27.	Turbine First Stage Pressure	S	R	M	
28.	Emergency Plan Radiation Instruments	M	R	M	
29.	Environmental Monitors	M	NA	NA	
30.	Loss of Voltage/Degraded Voltage 480 Volt Safeguards Bus	NA	R	M	
31.	Trip of Main Feedwater Pumps	NA	NA	R	
32.	Steam Flow	S	R	M	
33.	T_{AVG}	S	R	M	
34.	Chlorine Detector, Control Room Air Intake*	NA	R	M	
35.	Ammonia Detector, Control Room Air Intake*	NA	R	M	
36.	Radiation Detectors, Control Room Air Intake*	NA	R	M	

*Effective January 1, 1985

TABLE 4.1-3

Accident Monitoring Instrumentation Surveillance Requirements

Instrument	Channel Check	Channel Calibration	Channel Test
1. Pressurizer Water Level (1)	see Table 4.1-1	see Table 4.1-1	NA
2. Auxiliary Feedwater Flow Rate (4)	see Section 4.8.1	R	NA
3. Reactor Coolant System Subcooling Margin Monitor (2)	M	R	NA
4. Pressurizer PORV Position Indicator (primary detector) (3)	M	NA	R
5. Pressurizer PORV Position Indicator (RTD - backup detector) (3)	M	R	NA
6. PORV Block Valve Position Indicator (1)	M	NA	R
7. Pressurizer Safety Valve Position Indicator (primary detector) (3)	M	R	NA
8. Presurizer Safety Valve Position Indicator (RTD - backup detector) (3)	M	R	NA
9. Containment Pressure	M	R	NA
10. Steam Generator Water Level - Wide Range	M	R	NA
11. Containment Water Level (Narrow Range, Sump A)	M	R	NA
12. Containment Water Level (Wide Range, Sump B)	M	R	NA
13. Core Exit Thermocouples	M	R	NA
14. Containment Area High Range Radiation (R-29 and R-30) (5)	M	R	NA
15. Plant Vent (R-14A)	M	R	NA
16. A Main Steam Line (R-31)	M	R	NA
17. B Main Steam Line (R-32)	M	R	NA
18. Containment Purge (R-12A)	M	R	NA
19. Air Ejector (R-15A)	M	R	NA

(1) Emergency Power Supply Requirements for Pressurizer Level Indicators - NUREG 0578 Item 2.1.1
(2) Instrumentation for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.1
(3) Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 item 2.1.3.a
(4) Auxiliary Feedwater Flow Indication to Steam Generator NUREG 0578 item 2.1.7.b
(5) Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG 0737

a) to demonstrate a reactor coolant loop operable, the reactor coolant pump(s), if not in operation, shall be demonstrated operable at least once per 7 days by verifying correct breaker alignments and indicated power availability.

b) to demonstrate a residual heat removal pump is operable, the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a shall be performed.

4.3.5.4 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F and fuel is in the reactor, at least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.5 In addition to the above requirements, in order to demonstrate that a reactor coolant loop is operable, the steam generator water level shall be greater than or equal to 16% of the narrow range instrument span.

4.3.5.6 Each reactor coolant system vent path shall be demonstrated operable at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Verifying flow through the reactor coolant vent system vent paths using either liquid or gas.

<u>Basis</u>:

This material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of the reactor resulting from exposure to

neutron irradiation and the thermal environment. The test data obtained from this program will be used to determine the conditions under which the reactor vessel can be operated with adequate margins of safety against fracture throughout its service life.

The surveillance requirements on pressurizer equipment will assure proper performance of the pressurizer function and give early indication of malfunctions.

the tendon containing 6 broken wires) shall be inspected. The acceptance criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

4.4.4.2 Pre-Stress Confirmation Test

a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.

b. Before reseating a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

4.4.5.1 Each isolation valve specified in Table 3.6-1 shall be demonstrated to be operable in accordance with the Ginna Station Pump and Valve Test Program submitted in accordance with 10 CFR 50.55a.

4.4.6 Containment Isolation Response

4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.4.6.2 The RESPONSE TIME of each containment isolation function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific function as shown in the "Total No. of Channels" Column of Table 3.5-3. The response time limit shown on Table 3.6-1 does not include diesel generator starting times but does include valve travel times for all valves that change position. The times determined in independent tests, such as electronic response of portions of the initiating circuitry and valve travel times, may be combined to determine the total function response time.

4.4.7 Containment Hydrogen Monitors

4.4.7.1 Demonstrate that two hydrogen monitors are operable at least daily by verifying that the unit is on or in standby.

4.4.7.2 At least once per quarter perform a channel calibration using two sample gases containing known concentrations of hydrogen.

Basis:

The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286°F.

Attachment B

Licensees were requested to propose additional TMI Technical Specifications by Generic Letter 83-37. Enclosure 1 to that letter detailed eleven topics that were to be addressed in the plant technical specifications. Each of these topics has been considered as outlined below.

(1) Reactor Coolant System Vents (II.B.1)

New specifications 3.1.1.6 and 4.3.5.6 are proposed which require operability and surveillance testing of this equipment. Consistent with our current approved practice, valve cycling requirements will be included in the Ginna ISI program in Appendix C of the Quality Assurance Manual.

(2) Post Accident Sampling (II.B.3)

The post accident sampling capability required by NUREG-0737 has been provided. Existing technical specification 6.8.1.a requires that applicable procedures recommended by Regulatory Guide 1.33 be established, implemented and maintained. This requires that procedures exist to respond to emergency conditions and significant events, for control of radioactivity including sampling and monitoring and for control of measuring and test equipment. In addition, technical specification 6.8.1.e requires that procedures be established for implementation of the Emergency Plan. Additional specifications are unnecessary and will only add to the already sizable volume of specifications and place an additional burden on operators and those required to know the technical specifications. Plant readiness to cope with emergency conditions is periodically audited by NRC Inspection and Enforcement.

(3) Long-Term Auxiliary Feedwater System Evaluation (II.E.1.1)

Post TMI auxiliary feedwater system specifications were included in Amendment 42 and are found in sections 3.4 and 4.8. Additional correspondence related to this system is found in references 4 through 20 of a letter from John E. Maier to Dennis M. Crutchfield, USNRC, dated January 19, 1982. Alignment of safeguards systems was also addressed by RG&E letter dated May 22, 1980 from L. D. White, Jr. to Dennis M. Crutchfield and was approved by a letter from Mr. Crutchfield dated June 16, 1982.

(4) Noble Gas Effluent Monitors (II.F.1)

Operability of the post accident noble gas effluent monitors is required by proposed specification 3.5.2.4. Surveillance of these monitors is required by an addition to existing technical specification Table 4.1-3.

(5) Sampling and Analysis of Plant Effluents (II.F.1.2)

Additional specifications are unnecessary as described in (2) above. The existing technical specifications along with the specifications proposed in (4) above assure that the required capability exists.

(6) Containment High Range Radiation Monitor (II.F.1.3)

Containment area high range radiation monitor operability and surveillance requirements are given in proposed specification 3.5.2.4 and Table 4.1-3.

(7) Containment Pressure Monitor (II.F.1.4)

Containment pressure monitor operability and surveillance requirements have been added in proposed revisions to Tables 3.5-4 and 4.1-3.

(8) Containment Water Level Monitor (II.F.1.5)

Containment pressure monitor operability and surveillance requirements have been added in proposed revisions to Tables 3.5-4 and 4.1-3.

(9) Containment Hydrogen Monitor (II.F.1.6)

Proposed specifications 3.6.4 and 4.4.7 include operability and surveillance requirements for the containment hydrogen monitors.

(10) Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

Operability and surveillance requirements for the subcooling margin monitors and core exit thermocouples are included in revised Tables 3.5-4 and 4.1-3. RG&E has committed to install a differential pressure type reactor vessel water level instrument by 1986. Appropriate technical specifications will be proposed for the water level instrument after its installation and testing is complete.

(11) Control Room Habitability Requirements (III.D.3.4)

Control room HVAC detection systems operability and surveillance requirements are included in proposed specification 3.5.5 and revised Table 4.1-1. Approval

of the existing plant configuration to meet control room habitability requirements was issued in an NRC letter from Dennis M. Crutchfield to John E. Maier dated April 11, 1983. Operability and surveillance requirements for the control room emergency air treatment system are given in existing technical specifications 3.3.5 and 4.5.2.3.6 through 4.5.2.3.9.

In accordance with 10 CFR 50.91, the proposed changes to the Technical Specifications have been evaluated against three criteria to determine if the operation of the facility in accordance with the proposed amendment would:

1. involve a significant increase in the probability or consequences of an accident previously evaluated; or

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

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

None of the proposed changes will have an adverse impact as judged against these criteria.

Each of the proposed changes noted above constitute an additional limitation, restriction or control not presently included in the technical specifications or incorporates an additional surveillance requirement. These proposed changes thus conform to the Commission's example (ii) of changes that do not involve a significant hazards consideration.

In addition to the proposed changes addressed above, minor revisions have also been made which are administrative in nature. Table 3.5-4 has been revised to reference the requirements of NUREG-0737 instead of NUREG-0578, to correctly describe pressurizer discharge pipe temperature elements as RTDs instead of thermocouples, and to reflect that auxiliary feedwater flow indication is required for each steam generator instead of each pump. A note has been deleted from Table 4.1-1 which required that the loss of voltage/degraded voltage surveillance specification be effective no later than the Spring 1982 refueling outage. These administrative changes conform to the Commission's example (i) of changes that do not involve a significant hazards consideration.

Therefore, a no significant hazards consideration is warranted for the proposed technical specification changes.