3. The requirements of Specification 3.0.1 and 6.6.2 are not applicable.

F. The accident monitoring instrumentation for its associated operable components listed in TS Table 3.7-6 shall be operable in accordance with the following:

1. With the number of operable accident monitoring instrumentation channels less than the total number of channels shown in TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 7 days or be in at least hot shutdown within the next 12 hours.

2. With the number of operable accident monitoring instrumentation channels less than the minimum channels operable requirement of TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

G. The containment hydrogen analyzers and associated support equipment shall be operable in accordance with the following:

1. A reactor shall not be made critical nor be operated at power without two independent containment hydrogen analyzers operable.

2. During power operation or return to criticality from hot shutdown conditions, the following restrictions apply:



|| Designates Technical Specification change submitted to NRC on 5-4-83.

a. With one hydrogen analyzer inoperable, restore the inoperable analyzer to operable status within 30 days or be in at least hot standby within the next 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one analyzer to operable status within 7 days or be in at least hot standby within the next 6 hours.

Note: Operability of the hydrogen analyzers includes proper operation of the respective Heat Tracing System.

monitor indication. The pressurizer safety valves utilize an acoustic monitor channel and a downstream high temperature indication channel. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations". Potential accident effluent release paths are equipped with radiation monitors to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. The effluent release paths monitored are the Process Vent Stack, Ventilation Vent Stack, Main Steam Safety Valve and Atmospheric Dump Valve discharge and the Auxiliary Feedwater Pump Turbine Exhaust. These monitors meet the requirements of NUREG 0737.

Radioactive Liquid Effluent Monitoring Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/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 purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to unrestricted areas.

|| Designates Technical Specification change submitted to NRC on 5-4-83.

Radioactive Gaseous Effluent Monitoring Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) 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 Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Containment Hydrogen Analyzers

Continuous indication of hydrogen concentration in the containment atmosphere is provided in the control room over the range of 0 to 10 percent hydrogen concentration.

These redundant, qualified hydrogen analyzers are shared by Units 1 and 2 with the capability of measuring containment hydrogen concentration for the range of 0 to 10 percent and the installation of instrumentation to indicate and record this measurement.

A transfer switch with control circuitry is provided for the capability of Unit 1 to utilize both analyzers or for Unit 2 to utilize both analyzers.

Each unit's hydrogen analyzer will receive a transferable power supply from Unit 1 and Unit 2. This will ensure redundancy for each unit.

|| Designates Technical Specification change submitted to NRC on 5-4-83

Indication of Unit 1 and Unit 2 hydrogen concentration is provided on Unit 1 PAMC panel and Unit 2 PAMC panel. Hydrogen concentration is also recorded on qualified recorders. In addition, each hydrogen analyzer is provided with an alarm for trouble/high hydrogen content. These alarms are located in the control room.

The supply lines installed from the containment penetrations to the hydrogen analyzers have Category I Class IE heat tracing applied. The heat tracing system receives the same transferrable emergency power as is provided to the containment hydrogen analyzers. The heat trace system is de-energized during normal system operation. Upon receipt of a safety injection signal (Train A or Train B), the system is automatically started, after a preset time delay, to bring the piping process temperature to 250°F ± 10°F within 20 minutes. Each heat trace circuit is equipped with an RTD to provide individual circuit readout, over temperature alarm and cycles the circuit to maintain the process temperature via the solid state control modules.

The hydrogen analyzer heat trace system is equipped with high temperature, loss of D. C. power, loss of A. C. power, loss of control power and failure of automatic initiation alarms.

References

(1) FSAR - Section 7.5

(2) FSAR - Section 14.5

(3) FSAR - Section 14.3.2

(4) FSAR - Section 11.3.3

## TABLE 3.7-5

## AUTOMATIC FUNCTIONS OPERATED FROM RADIATION MONITORS ALARM

| | MONITOR CHANNEL | AUTOMATIC FUNCTION AT ALARM CONDITIONS | MONITORING REQUIREMENTS | ALARM SETPOINT µCI/cc |
|---|---|---|---|---|
| 1. | Process vent particulate and gas monitors (RM-GW-101 & RM-GW-102) | Stops discharge from containment vacuum systems and waste gas decay tanks (shuts Valve Nos. RCV-GW-160, FCV-GW-260, FCV-GW-101) | See Specifications 3.11 and 4.9 | Particulate $\leq 4x10^{-8}$<br>Gas $\leq 9x10^{-2}$ |
| 2. | Component cooling water radiation monitors | Shuts surge tank vent valve HCV-CC-100 | See Specifications 3.13 and 4.9 | Twice Background |
| 3. | Liquid waste disposal radiation monitors (RM-LW-108) | Shuts effluent discharge valves FCV-LW-104A and FCV-LW-104B | See Specification 3.11 and 4.9 | $\leq 1.5x10^{-3}$ |
| 4. | Condenser air ejector radiation monitors (RM-SV-111 & RM-SV-211) | Diverts flow to the containment of the affected unit (Opens TV-SV-102 and shuts TV-SV-103 or opens TV-SV-202 and shuts TV-SV-203) | See Specification 3.11 and 4.9 | $\leq 1.3$ |
| 5. | Containment particulate and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260) | Trips affected unit's purge supply and exhaust fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D) | See Specifications 3.10 and 4.0 | Particulate $\leq 9x10^{-9}$<br>Gas $\leq 1x10^{-5}$ |
| 6. | Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262) | Trips affected unit's purge supply and exhaust fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D) | See Specifications 3.10 and 4.9 | $\leq$ 50 mrem/hr |
| 7. | Process vent normal and high range effluent monitors (RM-GW-130-1 and RM-GW-130-2) | Stops discharge from containment vacuum system and waste gas decay tanks (shuts valves FCV-GW-160, FCV-GW-260, and FCV-GW-101) | See Specifications 3.11 and 4.9 | Gas $\leq 9x10^{-2}$ |

TABLE 3.7-6
ACCIDENT MONITORING INSTRUMENTATION

| | INSTRUMENT | TOTAL NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|---|
| 1. | Auxiliary Feedwater Flow Rate | 1 per S/G | 1 per S/G |
| 2. | Reactor Coolant System Subcooling Margin Monitor | 2 | 1 |
| 3. | PORV Position Indicator (Primary Detector) | 1/valve | 1/valve |
| 4. | PORV Position Indicator (Backup Detector) | 1/valve | 0 |
| 5. | PORV Block Valve Position Indicator | 1/valve | 1/valve |
| 6. | Safety Valve Position Indicator (Primary Detector) | 1/valve | 1/valve |
| 7. | Safety Valve Position Indicator (Backup Detector) | 1/valve | 0 |
| 8. | Reactor Vessel Coolant Level Monitor | 2 | 1 |
| 9. | Containment Pressure | 2 | 1 |
| 10. | Containment Water Level (Narrow Range) | 2 | 1 |
| 11. | Containment Water Level (Wide Range) | 2 | 1 |
| 12. | Containment High Range Radiation Monitor | 2 | 1 (Note 1, b and c only) |
| 13. | Process Vent High Range Effluent Monitor | 2 | 2 (Note 1, a, b, and c) |
| 14. | Ventilation Vent High Range Effluent Monitor | 2 | 2 (Note 1, a, b, and c) |
| 15. | Main Steam High Range Radiation Monitors (Units 1 and 2) | 3 | 3 (Note 1, a, b, and c) |
| 16. | Aux. Feed Pump Steam Turbine Exhaust Radiation Monitor | 1 | 1 (Note 1, a, b, and c) |

Note 1: With the number of operable channels less than required by the Minimum Channels Operable requirements

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours
b. Either restore the inoperable channel to operable status within 7 days of the event, or
c. Prepare and submit a Special Report to the commission pursuant to specification 6.6 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable.

|| Designates Technical Specification change submitted to NRC on 3-31-83.

# 4.1 OPERATIONAL SAFETY REVIEW

## Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

## Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

## Specification

A. Calibration, testing, and checking of instrumentation channels shall be performed as detailed in Table 4.1-1 and 4.1-2.

B. Equipment tests shall be conducted as detailed below and in Table 4.1-2A.

1. Each Pressurizer PORV shall be demonstrated operable:
   a. At least once per 31 days by performance of a channel functional test, excluding valve operation, and
   b. At least once per 18 months by performance of a channel calibration.

2. Each Pressurizer PORV block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel.

TABLE 4.1-2

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|
| 1. Auxiliary Feedwater Flow Rate | P | R |
| 2. Reactor Coolant System Subcooling Margin Monitor | M | R |
| 3. PORV Position Indicator (Primary Detector) | M | R |
| 4. PORV Position Indicator (Backup Detector) | M | R |
| 5. PORV Block Valve Position Indicator | M | R |
| 6. Safety Valve Position Indicator | M | R |
| 7. Safety Valve Position Indicator (Backup Detector) | M | R |
| 8. Reactor Vessel Coolant Level Monitor | M | R |
| 9. Containment Pressure | M | R |
| 10. Containment Water Level (Narrow Range) | M | R |
| 11. Containment Water Level (Wide Range) | M | R |

|| Technical Specification change submitted to NRC on 3-31-83.

TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

| | DESCRIPTION | TEST | FREQUENCY | FSAR SECTION REFERENCE |
|---|---|---|---|---|
| 18. | Primary Coolant System | Functional | 1. Periodic leakage testing [a] on each valve listed in Specification 3.1.C.7a shall be accomplished prior to entering power operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. | |
| 19. | Containment Purge MOV Leakage | Functional | Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval (c) | |
| 20. | Containment Hydrogen Analyzers | a. Channel Check | Once per 12 hours | |
| | | b. Channel Functional Test | Once per 31 days | |
| | | c. Channel Calibration using sample gas containing: 1. One volume percent (± 0.25%) hydrogen, balance nitrogen 2. Four volume percent (± 0.25%) hydrogen, balance nitrogen 3. Channel Calibration test will include startup and operation of the Heat Tracing System | Once per 92 days on staggered basis | |

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

*See Specification 4.1.D.

ATTACHMENT 2

DISCUSSION OF PROPOSED TECHNICAL SPECIFICATION CHANGE

NUREG-0737 requires the submittal of Technical Specifications for certain additional Accident Monitoring Instrumentation Items: II.F.1.1 - Noble Gas Effluent Monitors, II.F.1.3 - Containment High Range Radiation Monitors, II.F.1.4 - Containment Pressure Monitors, II.F.1.5 - Containment Water Level Monitors, and II.F.1.6 - Containment Hydrogen Monitors. The NRC required effluent monitors of NUREG-0737 have been installed, tested and are calibrated except for the Containment High Range Radiation Monitors. These monitors still require a final in-situ calibration.

A brief description of each additional Accident Monitoring Instrumentation follows:

II.F.1.1

Noble gas effluent monitors with an upper range capacity of $10^5$ uCi/cc (Xe-133) have been installed. The monitors have the capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. The monitors have a digital readout in the control room to provide the operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

II.F.1.3

Containment High-Range Radiation Monitors, with a maximum range of $10^7$ R/hr have been installed. The monitors provide the capability to detect and measure the radiation level within the reactor containment during and following an accident. The two high-range monitors are placed in separate areas of the containment to provide independent measurements and will "view" a large fraction of containment volume.

II.F.1.4

Containment Pressure Monitors that provide a continuous indication in the control room of containment pressure have been installed. Measurement and indication capability ranges from three times the design pressure of the containment to 5 psia.

II.F.1.5

Containment Water Level Monitors that provide continuous indication of containment water level have been installed and the monitors have a readout in the control room. A narrow range monitor is installed to cover the range of water from the bottom to the top of the containment sump. The wide range monitor is installed to cover the range of water from the bottom of the containment to the equivalent to a 600,000 gallon capacity.

II.F.1.6

Containment Hydrogen Monitors that provide indication of hydrogen concentration in the containment atmosphere has been installed and indication is provided in the control room. Measurement capability is provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure. The containment isolation valves associated with the Containment Hydrogen Monitors in the Hydrogen Analyzer System were added to Tables 3.8-1 and 3.8-2 in a previous Technical Specification change which was submitted to the NRC on January 10, 1983.

Item II.F.1.1, the noble gas effluent monitors, more specifically, the Main Steam High Range Radiation Monitor and the Auxiliary Feed Pump Steam Turbine High Range Radiation Monitors and Item II.F.1.3, Containment High Range Radiation Monitors are considered part of the Processing and Area Radiation Monitoring Systems. As such, the surveillance requirements of Items II.F.1.1 and II.F.1.3 are covered in Table 4.1-1 of the Technical Specifications under Channel Description 19.

The addition of additional Accident Monitoring Instrumentation (NUREG-0737, II.F.1) increases the overall plant margin of safety by providing monitoring of potential radioactive release paths and provides a means to monitor hydrogen buildup in containment, radiation levels in containment, and containment pressure and water levels. In no case is the information provided by these monitoring systems used to initiate automatic activation of any plant safety systems.

This proposed change does not involve a significant hazards consideration. The probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR is not increased, nor has the possibility of an accident or malfunction of a different type previously evaluated in the UFSAR been created do to the additional accident monitoring instrumentation. The margin of safety as defined in the Basis for any Technical Specification is not reduced because the operation or failure of these systems does not lessen any safety system's ability to perform its design function. These systems are used during and following an accident therefore providing information related to plant status and condition.

ATTACHMENT 3

c. With the pressurizer otherwise inoperable, be in at least hot shutdown with the reactor trip breakers open within 6 hours and the reactor coolant system temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours.

6. Relief Valves

a. Two power operated relief valves (PORVs) and their associated block valves shall be operable whenever the reactor keff is ≥ 0.99.

b. With one or more PORVs inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

c. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

7. Reactor Vessel Head Vents

a. At least two Reactor Vessel Head vent paths consisting of two isolation valves in series powered from emergency buses shall be operable and closed whenever RCS temperature and pressure are > 350°F and 450 psig.

b. With one Reactor Vessel Head vent path inoperable; startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of both isolation valves in the inoperable vent path.

c. With two Reactor Vessel Head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuator of all isolation valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 30 days or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

Basis

Specification 3.1.A-1 requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling flow in the event of a loss of reactor coolant flow accident. This provided flow will maintain the DNBR above 1.30.(1) Heat transfer analyses also show that reactor heat equivalent to approximately 10% of rated power can be removed with natural circulation; however, the plant is not designed for critical operation with natural circulation or one loop operation and will not be operated under these conditions.

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 maintain a uniform concentration 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 equivalent of the reactor coolant system volume in approximately one half hour.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The requirement for redundant coolant loops ensures the capability to remove core decay heat when the reactor coolant system average temperature is less than or equal to 350°F. Because of the low-low steam generator water level reactor trip, normal reactor criticality cannot be achieved without water in the steam generators in reactor coolant loops with open loop stop valves. The requirement for two operable steam generators, combined with the requirements of Specification 3.6, ensure adequate heat removal capabilities for reactor coolant system temperatures of greater than 350°F.

Each of the pressurizer safety valves is designed to relieve 295,000 lbs. per hr. of saturated steam at the valve setpoint. Below 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. There are no credible accidents which could occur when the Reactor Coolant System is connected to the Residual Heat Removal System which could give a surge rate exceeding the capacity of one pressurizer safety valve. Also, two safety valves have a capacity greater than the maximum surge rate resulting from complete loss of load. [2]

The limitation specified in item 4 above on reactor coolant loop isolation will prevent an accidental isolation of all the loops which would eliminate the capability of dissipating core decay heat when the Reactor Coolant System is not connected to the Residual Heat Removal System.

The requirement for steam bubble formation in the pressurizer when the reactor passes 1% subcriticality will ensure that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that 125 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.

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 accumulation of non-condensable gases in the Reactor Coolant System may result from sudden depressurization, accumulator discharges and/or inadequate core cooling conditions. The function of the Reactor Vessel Head Vent is to remove non-condensable gases from the reactor vessel head. The Reactor Vessel Head Vent is designed with redundant safety grade vent paths. Venting of non-condensable gases from the pressurizer steam space is provided primarily through the Pressurizer PORVs. The pressurizer is, however, equipped with a steam space vent designed with redundant safety grade vent paths.

References:

(1) FSAR Section 14.2.9

(2) FSAR Section 14.2.10

ATTACHMENT 4

3. The pressurizer water volume shall be determined to be within its limit as defined in Specification 2.3.A.3.a at least once per 12 hours whenever the reactor is not subcritical by at least 1% Δk/k.

4. Each Reactor Vessel Head vent path isolation valve not required to be closed by Specification 3.1.A.7a or 3.1.A.7.b shall be demonstrated operable at each cold shutdown but not more often than once per 92 days by operating the valve through one complete cycle of full travel from the control room.

5. Each Reactor Vessel Head vent path shall be demonstrated operable following each refueling by:

   a. Verifying that the upstream manual isolation valve in each vent path is locked in the open position.

   b. Cycling each isolation valve through at least one complete cycle of full travel from the control room.

   c. Verifying flow through the reactor vessel head vent system vent paths.

C. Sampling tests shall be conducted as detailed in Table 4.1-2B.

D. Whenever containment integrity is not required, only the asterisked items in Table 4.1-1 and 4.1-2A and 4.1-2B are applicable.

E. Flushing of sensitized stainless steel pipe sections shall be conducted as detailed in TS Table 4.1-3A and 4.1-3B.

F. The outside containment purge and vent isolation valves and the isolation valve in the steam jet air ejector suction line outside containment shall be determined locked, sealed, or otherwise secured in the closed position at least once per 31 days.

G. The inside containment purge and vent isolation valves shall be determined locked, seal, or otherwise secured in the closed position each cold shutdown but not more often than once per 92 days.

DISCUSSION OF PROPOSED TECHNICAL SPECIFICATION CHANGE

NUREG-0737, Item II.B.1, required the installation of the Reactor Vessel Head Vent (RVHV) whose function is to remove non-condensable gases from the reactor vessel head. The Reactor Vessel Head Vent is designed with redundant safety grade vent paths.

The Reactor Vessel Head Vent System will extend the reactor coolant pressure boundary to and including the 3/8 in. orifices in the Reactor Vessel Head Vent. The system is designed, fabricated, and installed in accordance with the requirements of FSAR Section 5, and all applicable codes, as part of the Reactor Coolant System.

Loss of reactor coolant resulting from vent failure is categorized as being a loss of coolant accident which is fully bounded by previous evaluations, while a failure downstream of the 3/8 in. orifices is within the capacity of the normal reactor coolant makeup system. System design provides for manual initiation or termination of venting effective with the single failure criteria.

This proposed change does not involve a significant hazards consideration. This system does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Safety Analysis Report, nor does it create the possibility of an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report. The margin of safety as defined in the basis for any technical specification is not reduced, and this system does not lessen any other system's ability to perform its design function.