Bases Continued:

2.3 The operator will set the low low water ECCS initiation trip setting 6'6" 6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page 18.

E. Turbine Control Valve Fast Closure Scram - The turbine control valve fast closure scram and PRT is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Reference Sections 14.5.1.1 and 14.5.1.2 FSAR and supplemental information entitled "Permanent Plant Changes to Accommodate Equilibrium Core Scram Reactivity Insertion Characteristics", dated January 23, 1974.

F. Turbine Stop Valve Scram - The turbine stop valve scram with PRT, like the load rejection scram with PRT, anticipates the pressure, neutron flux and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a setting at 10% of valve closure for scram, and PRT, the increase in heat flux is limited such that adequate pressure and thermal margins are maintained. The PRT opens safety/relief valves to limit the pressure and heat flux increases, and allows safety/relief valve reclosure as pressure decreases. For this event, the peak surface heat flux and MCHFR remain within limits. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted January 23, 1974.

G. Main Steam Line Isolation Valve Closure Scram - The main steam line isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 850 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 18.

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor vessel pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel

2.4 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be $\leq$ 1075 psig.

B. Reactor Coolant System Safety/Relief Valves shall be set as follows:

6 valves at $\leq$ 1080 psig.

Bases Continued:

2.2 The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The peak pressure in this transient is limited to 1178 psig at the bottom of the vessel. The primary system overpressure protection analysis assumes the closure of all MSIV's with indirect (high flux) scram. Peak pressure at the vessel bottom is 1285 psig.

Bases:

2.4 The settings on the reactor high pressure scram, prompt relief trip system, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety/relief valves offer yet another protective feature for the reactor coolant system pressure safety limit. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the code for a condition of MSIV closure while operating at 1670 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1285 psig. See FSAR Section 4.4.3 and supplemental information submitted January 23, 1974. Evaluations presented indicate that a total of six dual purpose safety/relief valves set at 1080 psig maintain the peak pressure during the transient within the limits allowed by the ASME Code.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1080 psig or lower. However, the actual set point can be as much as 11 psi above the 1080 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. Analyses were performed assuming a safety/relief valve setpoint of 1080 psig + 1%.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or when a sufficient number of devices have been affected by any means

Bases Continued:

3.1 condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. The turbine stop valve closure scram with PRT adequately preserves the margins to pressure and MCHFR limits should a turbine trip with bypass failure occur. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted January 23, 1974. The condenser low vacuum scram is a back-up to stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds ten times normal full power background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive release of radioactive materials. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation to the main condenser off-gas line provided the instantaneous limit specified in Specification 3.8 is exceeded for a 15-minute period.

The main steamline isolation valve closure scram is set to scram when the isolation valves are ≤10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. Reference Section 14.5.1.3.1 FSAR and supplemental information submitted February 13, 1973.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.7.1 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 IRM system provides protection against excessive power levels and short reactor periods in the

E. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

1. a. Except as specified in 3.2.E.1.b below, four radiation monitors shall be operable at all times.

   b. One of the two monitors in the ventilation plenum and one of the two radiation monitors on the refueling floor may be inoperable for 24 hours. If the inoperable monitors are not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment system operated until repairs are complete.

2. The radiation monitors shall be set to trip as follows:

   (a) ventilation plenum $\leq$ 3 mr/hr
   (b) refueling floor $\leq$100 mr/hr

3. When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212^{o}F, the limiting conditions for operation for the instrumentation listed in Table 3.2.4 shall be met.

F. Prompt Relief Trip System

The limiting conditions of Operation for the instrumentation that initiates prompt relief trip are given in Table 3.2.4.A.

TABLE 3.2.4A

Instrumentation that Initiates Prompt Relief Trip

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2,3)	Required Conditions*
1. Turbine Stop Valve Closure	Note [illegible]	2	2	A
2. Turbine Control Valve Fast Closure	Note 4	2	2	A
3. Reactor Low Pressure - PRT Disable	$\geq$900 $\leq$ 950 psig	2	2	A
4. PRT Timer	3-8 sec	2	2	A
5. Turbine First Stage Pressure (% of Rated)	$\leq$70% $\leq$85%	1 1	1 1	A (Note 5) B (Note 5)

Notes

1. There shall be two operable or operating trip systems for each function.

2. One instrument channel may be bypassed to permit testing of the channel.

3. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated to:

   (a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or
   (b) place the plant under the specified required conditions using normal operating procedures.

TABLE 3.2.4A (Continued)

Notes (Continued)

4. PRT initiating signals operate directly from scram initiating instruments. Trip settings are previously established in Table 3.1.1.

5. The PRT system consists of two, two-channel three-valve subsystems. System functional requirements include one three-valve subsystem operable for power levels greater than seventy percent and the other three-valve subsystem operable for power levels greater than eighty-five percent. Should a component of one of the subsystems become inoperable, reactor power will be reduced to the corresponding power level within 24 hours until repairs are implemented. All installed safety/relief valves are available for PRT service. However, only six are actively engaged at any one time. Should one of the six become inoperable, a spare valve can be activated in its place without shutting down.

* Required conditions when minimum conditions for operation are not satisfied.

A. Reactor power less than 70% of rated
B. Reactor power less than 85% of rated

Table 4.2.1 - Continued

Minimum Test and Calibration Frequency For Core Cooling, Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Note 1	Once/3 months	None
4. Steam Line High Radiation	Once/week (5)	Note 6	Once/shift
HPCI ISOLATION			
1. Steam Line High Flow	Note 1	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once/3 months	None
RCIC ISOLATION			
1. Steam Line High Flow	Note 1	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once 3/months	None
REACTOR BUILDING VENTILATION			
1. Radiation Monitors (Plenum)	Note 1	Once/3 months	Once/shift
2. Radiation Monitors (Refueling Floor)	Note 1	Once/3 months	(4)
OFF-GAS ISOLATION			
1. Radiation Monitors (Air Ejectors)	Notes (1,5)	Note 6	Once/shift
PROMPT RELIEF TRIP (PRT) SYSTEM			
1. PRT Disable (Reactor Low Pressure)	Note 1	Once/3 months	None
2. PPT Power Range Permissive (Turbine First Stage Pressure)	Note 1	Once/3months	None
3. PRT Timer	Note 1	Once/3 months	None
4. Turbine Stop Valve Closure	Note 7	None	None
5. Turbine Control Valve Closure	Note 7	Note 7	None

NOTES:

(1) Initially once per month until exposure hours (M as defined on Figures 4.1.1) is 2.0×10^5, thereafter according to Figure 4.1.1. with an interval not greater than three months.

Table 4.2.1 - Continued

NOTES:

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per shift and test once per week until no longer required.

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

(4) Whenever fuel handling is in process, a sensor check shall be performed once per shift.

(5) A Functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

(6) This instrument will be calibrated every three months by means of a built in current source, and each refueling outage with a known radioactive source.

(7) Instrument functional test and calibration shall include verification of instrument channel response in the PRT system. Frequencies are established in TS Tables 4.1.1 and 4.1.2.

Bases:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and the Prompt Relief System.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 10'6" (7" on the instrument at 100% rated thermal power) above the top of the active fuel. This trip initiates closure of Group 2, and 3 primary containment isolation valves. Reference Section 7.7.2.2 FSAR. For a trip setting of 10'6" above the top of the active fuel, the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 6'6" above the top of the active fuel. This trip initiates closure of the Group 1 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generator.

Bases Continued:

3.2 For effective emergency core cooling for the small pipe break the HPCI or Automatic Pressure Relief system must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria is met. Reference Section 6.2.4 and 6.2.6 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or two downscale. There is a [illegible]-minute delay before recombiner train inlet valve closure when the recombiners are in use and a 15 minute delay before off-gas isolation valve closure when the recombiners are bypassed in which the reactor operator may take corrective action. Both instruments are required for trip. The trip settings of the instruments are set so that the maximum stack release rate limit allowed by Specification 3.8.A.1 is not exceeded.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation plenum and on the refueling floor. Any one upscale trip will cause the desired action. Trip settings of 3 mR/hr for the monitors in the ventilation duct are based upon initiating normal ventilation isolation and Standby Gas Treatment System operation so as not to exceed the maximum release rate limit allowed by Specification 3.8.A.1 for the reactor building vent. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The prompt relief trip (PRT) system initiates the opening of three or six safety/relief valves at reactor power levels ≥70% and ≥85% respectively, with the occurrence of a turbine stop valve or control valve closure. The PRT initiating action originates in two independent channels, each capable of satisfying system requirements through redundant instruments. Interval timers with a low pressure back-up disable the PRT; self actuated pressure operation of the safety/relief valves remains unaffected.

Bases Continued:

3.2 The settings of the instruments provided in Table 3.2.4A ensure that pressure and thermal margins are maintained during the worst-case single-failure-caused abnormal operational transient, i.e., turbine trip with failure of the bypass valves. In addition, the PRT utilizes a 1 out of 2 logic system. In accordance with IEEE-279 an exception is taken to the minimum operating requirements to allow a short period of time, during which an instrument channel may be bypassed to allow for testing. For this logic the single failure protection is temporarily defeated.

Although the operator will set the set points within the trip settings specified in Tables 3.2.1, 3.2.2, 3.2.3, and 3.2.4, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts.

Table 3.2.5 - Continued
Trip Function and Deviations

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	Low-Low Reactor Water Level	-3 Inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale	-2/125 of Scale
	IRM Upscale	+2/125 of Scale
	APRM Downscale	-2/125 of Scale
	APRM Upscale	See Basis 2.3 - Page 24
	RBM Downscale	-2/125 of Scale
	RBM Upscale	Same as APRM Upscale
Instrumentation That Controls the Prompt Relief Trip (PRT) System	PRT Disable (Reactor Low Pressure)	± 1%
	PRT Timer	± 0.5 sec.
	PRT Power Range Permissive (Turbine First Stage Pressure)	± 3% of rated pressure

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.398
20	0.954
50	2.120
90	3.80

4.0 C. Scram Insertion Times

During each operation cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

Bases Continued 3.3 and 4.3:

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10% of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The consequences of a rod block monitor failure have been evaluated and reported in the Dresden II SAR Amendments 17 and 19. These evaluations, equally applicable to Monticello, show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Engineer, Nuclear, to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to ensure the maintenance of adequate fuel thermal and reactor pressure margins for non-accident events. This requires the negative reactivity insertion in any local region of the core and in the overall core to be at least as great as the (End-of-Cycle equilibrium core) scram reactivity insertion curve used in the analyses submitted on January 23, 1974. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity locally and in the overall core, respectively. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram as presented in the FSAR and the supplemental information submitted January 23, 1974. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCHFR remains greater than 1.35. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods.

Bases Continued:

margin, the RCIC system (a non-safeguard system) has been required to be operable during this time,
since the RCIC system is capable of supplying significant water makeup to the reactor (400 gpm).

E. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem.
They enable the core spray system or LPCI to provide protection against the small pipe break in
the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core
sprays or LPCI. Either of the two core spray systems or LPCI provide sufficient flow of coolant to
limit fuel clad temperatures to well below clad melt and to assure that core geometry remains intact.
Three safety/relief valves are included in the automatic pressure relief system. Of these three,
only two are required to provide sufficient capacity for the automatic pressure reliefsystem. See
section 4.4 and 6.2.5.3 FSAR.

F. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr. when averaged over a one-hour period.

2. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the idle recirculation loop is within 50°F of the reactor coolant temperature.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal Limitations

1. During heatups and cooldowns recirculation loops A and B temperatures shall be permanently recorded at 15 minutes intervals

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 minute intervals until three consecutive readings are within 5 degrees of each other.

(b) When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least once per shift and analyzed for conductivity and chloride ion content.

4. If Specification 3.6.C.1, 3.6.C.2, and 3.6.C.3 are not met, normal orderly shutdown shall be initiated.

D. Coolant Leakage

Any time irradiated fuel is in the reactor vessel, and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and have the reactor placed in the cold shutdown condition within 24 hours.

D. Coolant Leakage

Reactor coolant system leakage into the drywell shall be checked and recorded at least once per day.

E. Safety/Relief Valves and Prompt Relief Trip (PRT) System

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F.

   a. The safety valve function (self-actuation) of six safety/relief valves shall be operable.

   b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

2. During reactor power operation, the prompt relief trip (PRT) system function of six safety/relief valves shall be operable in accordance with Specification 3.2.F.

E. Safety/Relief Valves and Prompt Relief Trip (PRT) System

1. a. A minimum of six safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal setpoint of all operational safety/relief valves shall be $\leq$ 1080 psig.

   b. At least two of the safety relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

2. Surveillance of the PRT System shall be as follows:

Item	Frequency
Valve Operability	Each Operating Cycle
Simulated Automatic Actuation Test	Each Operating Cycle

3.0 LIMITING CONDITIONS FOR OPERATION

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

G. Jet Pumps

Whenever the reactor is in the Startup or Run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the plant shall be placed in a cold shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

F. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the Startup or Run modes, jet pump operability shall be checked daily by verifying that all the following conditions do not occur simultaneously:

1. The two recirculation loop flows are unbalanced by 15% or more when the recirculation pumps are operating at the same speed.
2. The indicated value of core flow rate is 10% or more less than the value derived from loop flow measurements.

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

An annual report will be prepared and submitted to the AEC summarizing the primary coolant to drywell leakage measurements. Other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety/Relief Valves and Prompt Relief Trip

Testing of all safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1080 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty through prompt relief trip and self actuation.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the mainsteam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 MW_t. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 71% of the full power steam generation rate.

The safety/relief valves have two functions; i.e. automatic vessel depressurization or over-pressure protection. The former is a solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate opening of the valves. This function provides backup to the HPCI system for small break protection and is discussed in Specification 3.5.E. In addition these valves can be operated manually.

The over-pressure protection function utilizes six safety/relief valves, three of which are operated for the Automatic Pressure Relief function. All six valves are capable of direct, self-actuation or indirect, prompt-relief trip (PRT) actuation.

The primary overpressure protection (for ASME Code consideration) is provided via the pressure-actuated integral bellows and pilot valve that cause main valve operation for any plant event wherein valve setpoint pressure is attained. Article 9 of the ASME Pressure Vessel Code Section III, Nuclear Vessels, requires that the bellows be monitored for failure since this would defeat the function of the safety/relief valve.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provides assurance of bellows integrity.

When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

F. Structural Integrity

A pre-service inspection of the components listed in Table 4.6.1 has been conducted to assure that the system is free of gross defects and as a reference base for later inspections. In addition, the facility has been designed such that gross defects should not occur throughout life. The inspection program was based on the proposed ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems which was followed except where accessibility for inspection was not provided. This inspection provides further assurance that gross defects are not occurring after the system is in service. This inspection will reveal problem areas should they occur before a leak develops.

Bases Continued 3.6 and 4.6:

Design confirmation and construction adequacy will be demonstrated during the plant startup and power ascension test program. As part of this program, cold and hot vibration tests on certain reactor vessel internals will be performed. The tests, described in a letter to Dr. P. A. Morris, dated March 5, 1970, are designed to obtain confirmatory data on the design features of Monticello as compared to Dresden Unit 2 design. Thus, the basis for the Monticello vibration test program is predicated on obtaining satisfactory data which confirms common design features from earlier BWR plants such as Dresden Unit 2. In the event that data from these earlier plants are not available before routine power operation of Monticello, the matter will be reviewed by the AEC.

The program outlined in Table 4.6.1 is limited to inspections of the primary coolant system. It is anticipated that the data collected during the first five years of operation will provide a suitable basis to evaluate the need for inspecting other portions of the facility (such as the main steam lines downstream of the main steamline isolation valves). These data along with the overall operating experiences will be reviewed to determine the inspection program to be implemented for the lifetime of the facility. The results of this study together with the proposed lifetime inspection program will be submitted to the AEC in accordance with Specification 6.7.C.3.

The special inspection of the main feed and steam lines is to provide added protection against pipe whip. The Group I welds are selected on the basis of an analysis that shows these welds are the highest stress welds and that due to their physical location, a break would result in the least interference and maximum energy upon impact with the drywell. These welds are the only ones which offer any significant risk and will be included in future inspections as determined by the study described above.

Group II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in the first inspection. Upon consideration of impact angle, interfering equipment and distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the AEC. These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

Bases Continued 3.6 and 4.6:

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

The prompt relief trip (PRT) function of the safety/relief valves provides an anticipatory actuation of the safety/relief valves for transients involving turbine stop valve closure or turbine control valve fast closure. Although the PRT system is intended to provide anticipatory pressure relief for turbine trip transients with failure of the bypass valves, the PRT system is entirely independent of the bypass system. The PRT system, by providing an anticipatory open signal to the safety/relief valves aids in maintaining fuel thermal and pressure margins and has therefore been designed on the basis of Engineered Safeguards and meets the requirements of IEEE 279.

The PRT system is divided into two independent redundant channels which are programmed on a power level schedule into three modes of operation relative to the safety/relief valves coupled to it. The modes of operation are listed below:

Reactor Power	PRT/S/RV's
85%	6
70%	3
70%	0

Mode selection is automatic through a biasing signal based on turbine first stage pressure. The power level schedule has been established to ensure the full power transients remain the most limiting and adequate pressure and fuel thermal margins are provided.

The PRT system will not preclude safety/relief valve self actuation or Automatic Pressure Relief Operation.

Deactivation of the PRT system signal is effected through redundant interval timers and a low reactor pressure switch or the attainment of the self-actuation reset pressure, should the pressure remain above the self-actuation setpoint for a period exceeding the interval timer setting.

Spare safety/relief valves may be installed that are adaptable to PRT service with minor interconnection changes to permit the substitution for an inoperable PRT valve while operating in a reduced power mode.

The PRT System is incorporated to offset changes in the scram reactivity insertion rate which occur with increasing exposure out to the equilibrium exposure. All transients have been analyzed to account for both the slower insertion rates and PRT installation in "Plant Changes to Accommodate Equilibrium Core Scram Reactivity Insertion Characteristics". This report was submitted to the Commission on January 23, 1974.

Bases Continued 3.6 and 4.6:

The PRT system demonstrates substantial improvement in fuel thermal and pressure response to the PRT-coupled turbine and generator trip transients. This effect is manifested in the results of a turbine trip with bypass failure transient where prt reduces peak vessel pressure by 74 psi and heat flux by 13% compared with the same event without PRT. PRT provides effective compensation for effects on plant performance caused by the changing scram reactivity. The PRT 3-mode design provides flexibility in maintaining pressure and fuel thermal margins and minimizing the duty cycle on the safety/relief valves at low power. Positive disabling of the PRT action is insured by a timer with a nominal setting of 5 seconds and a low vessel pressure signal set at 900-950 psig.

G. Jet pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within $\pm$ 5%, the flow rates in both recirculation loops will be verified by Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

6. If the specifications of 3.7.A cannot be met, the reactor shall be placed in a cold shutdown condition within 24 hours.

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both circuits of the standby gas treatment system shall be operable at all times when secondary containment integrity is required.

B. Standby Gas Treatment System

1. Standby gas treatment system surveillance shall be performed as indicated below:

   a. At least once per operating cycle it shall be demonstrated that:

      (1) Pressure drop across the combined high-efficiency and charcoal filters is less than 7.0 inches of water, and

      (2) Inlet heater output is at least 15 kw.

   b. During each refueling outage prior to refueling, whenever a filter is changed, whenever work is performed that could affect filter systems efficiency, and at intervals not to exceed six months between refueling outages, it shall be demonstrated that:

      (1) The removal efficiency of the installed particulate filters for particles having a mean diameter of 0.7 microns shall be

2. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit including its emergency power source shall be operable.

3. If this condition cannot be met, procedures shall be initiated immediately to establish the conditions listed in 3.7.C.1. (a) through (d), and compliance shall be completed within 24 hours thereafter.

equal to or greater than 99% based on an in-place dioctyl phthalate (DOP) test.

(2) The removal efficiency of the charcoal filters is not less than 99% for freon based on a freon test.

c. At least once each five years removable charcoal cartridges shall be removed and adsorption shall be demonstrated.

d. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.

2. When one circuit of the standby gas treatment system becomes inoperable, the operable circuit including its emergency power source shall be demonstrated to be operable immediately. The operable circuit of the Standby Gas Treatment System shall be demonstrated to be operable daily thereafter.

Bases Continued:

4.7 High efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by in-place testing with DOP as testing medium. Individual filter units will be tested and certified to have a removal efficiency of equal to or greater than 99% for particles having a mean diameter of 0.3 microns at the time of purchase.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, retention capacity in terms of microcuries of iodine per gram of charcoal will be demonstrated. This will be done by removing small test cartridges of the same charcoal filter material. These cartridges complement the charcoal filter system and will be available for withdrawal and testing. These tests will normally be performed every five years unless filter efficiency seriously deteriorates. Since shelf lives greater than five years have been demonstrated, the test interval is reasonable.

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses suggest that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. However, for added margin the Technical Specifications require a valve closure time of not greater than 5 seconds.

The primary containment isolation valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. Reference Section 5.2.2.4.3 and Table 5-2-3 FSAR. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

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