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Bases 2.1:

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is no less than the values specified in Technical Specification 2.1.A. This limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection systems safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling. (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The concept of MCPR, as used in the GETAB/GEXL critical power analyses, is discussed in Reference 1.

A. Core Thermal Power Limit (Reactor Pressure > 800 psia and Core Flow > 10% of Rated.) Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. The Safety Limit (T.S.2.1.A) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the Operating MCPR Limit (T.S.3.11.C) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle.

Bases 2.1 (Continued):

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the MCPR Safety Limit would not produce boiling transition. Thus, although it is not required to establish the Safety Limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Monticello operated above the boiling transition for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the MCPR Safety Limit, operation is constrained to a maximum design linear heat generation rate for any fuel type in the core.

B. Core Thermal Power Limit (Reactor Pressure $\leq$800 psia or Core Flow $\leq$10% of Rated) At pressure below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and all core flows, this pressure differential is maintained in the bypass region of the core.

Bases 2.1 (Continued):

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of $28x10^3$ lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than $28x10^3$ lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at $28x10^3$ lbs/hr is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient Plant safety analyses have shown that the scrams initiated by exceeding safety system setting will assure that the Safety Limit of 2.1.A or 2.1.B will not be exceeded. Control rod scram times and safety systems settings are checked periodically to assure that a scram will proceed as analyzed. As a further check, the plant process computer will be used as a fast data-acquisition system, when available during a scram, to verify that the scram was initiated by the primary source signal. The computer is normally available for this function. However, it is recognized that the plant may operate without the computer in service, in which event the confirmatory data will not be available and the verification specified by 2.1.C will not be required. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. For this specification, when a scram is only accomplished by means of a backup feature of the plant design, a specific analysis is required to determine whether or not a Safety Limit has been violated. The concept of not approaching a Safety Limit, providing scram signals are operable, is supported by the extensive plant safety analysis.

D. Reactor Water Level (Shutdown Condition) During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

Bases 2.1 (Continued):

References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO 10958.

Bases 2.3:

The abnormal operational transients applicable to operation of the Monticello Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power level of 1670 MWt. The analyses were based upon plant operation in accordance with the operating map. The licensed maximum power level 1670 MWt represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model.

Bases 2.3 (Continued):

For analyses of the thermal consequences of the transients, the Operating MCPR Limit (T.S.3.11.C) is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Deviations from as-left settings of setpoints are expected due to inherent instrument error, operator setting error, drift of the setpoint, etc. Allowable deviations are assigned to the limiting safety system settings for this reason. The effect of settings being at their allowable deviation extreme is minimal with respect to that of the conservatisms discussed above. Although the operator will set the setpoints within the trip settings specified, the actual values of the various setpoints can vary from the specified trip setting by the allowable deviation.

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 such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. Sections 3.1 and 3.2 list the reactor modes in which the functions listed above are required.

A. <u>Neutron Flux Scram</u> The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1670 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Also, the flow biased neutron flux scram (specification 2.3.A.1) provides protection to the fuel safety limit in the unlikely event of a thermal-hydraulic instability.

Bases 2.3 (Continued):

Maximum Extended Load Line Limit Analyses have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). These analyses have not changed the allowed operation for single loop operation. The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Increased Core Flow analyses have been performed to allow operating at flows above 100% for powers equal to or less than 100% (See Core Operating Limit Report). The supporting analyses are discussed in General Electric NEDC-31778P report (Reference: Letter from NSP to NRC dated September 16, 1992).

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

B. Deleted

Bases 2.3 (Continued):

C. Reactor Low Water Level Scram The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual setpoint can be as much as 6 inches lower due to the deviations discussed on page 39.

D. Reactor Low Low Water Level ECCS Initiation Trip Point The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters; the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could prevent the ECCS components from meeting their criterion. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

The operator will set the low low water level 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 39.

E. Turbine Control Valve Fast Closure Scram The turbine control valve fast closure scram 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. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% turbine first stage pressure. This takes into account the possibility of 15% power being passed directly to the condenser through the bypass valves.

F. Turbine Stop Valve Scram The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% turbine first stage pressure. This takes into account the possibility of 15% power being passed directly to the condenser through the bypass valves.

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 closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Main Steam Line Low Pressure Initiates Main Steam Isolation Valve Closure The low pressure isolation of the main steam lines at 825 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 at steamline pressures lower than 825 psig requires

Bases 2.3 (Continued):

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 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 39.

Bases 2.2:

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured in the vessel steam space is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value was derived from the design pressures of the reactor pressure vessel, coolant piping, and recirculation pump casing. The respective design pressures are 1250 psig at 575°F, 1148 psig at 562°F, and 1400 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and the USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10 percent over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure (120% x 1148 = 1378 psig).

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 1.5 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 29,400 psi, still safely below the yield strength.

The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

Bases 2.2 (Continued):

The normal operating pressure of the reactor coolant system is approximately 1010 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 safety/relief valves (S/RV's) are sized assuming no direct scram during MSIV closure. The only scram assumed is from an indirect means (high flux). The analysis assumes that only seven of the eight S/RV's are operable and that they open at 1% over their setpoint with a 0.4 second delay. Reactor pressure remains below the 1375 psig ASME Code limit for the reactor vessel.

Bases 2.4:

The settings on the reactor high pressure scram, 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 assure that the reactor coolant system pressure safety limit is never reached. 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 remains below the 1375 psig ASME Code limit. Only seven of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 1% above their setpoint with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the design pressure of the HPCI and RCIC systems of 1120 psig.

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 3.1. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1120 psig or lower. However, the actual set point can be as much as 11.2 psi above the 1120 psig indicated set point due to the deviations discussed in the basis of Specification 3.6.

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 2.4 (Continued):

such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. The functions listed in this specification are required in all modes except cold shutdown.

Bases 4.0:

This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operations are met and will be performed during the periods when the Limiting Conditions for Operation are applicable.

A tolerance for performing surveillance activities beyond the nominal interval is provided to allow operational flexibility because of scheduling and performance considerations. The plant uses a fixed surveillance program that prevents repetitive addition of the allowable 25% extension. Each surveillance test is completed within plus or minus 25% of each scheduled fixed date. Scheduled dates are based on dividing each calendar year into four 13-week "surveillance" quarters consisting of 3 4-week "surveillance" months and one "catch-up" week. This method of scheduling permits certain tests always to be scheduled on certain days of the week.

The specification ensures that surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into a plant condition for which the Limiting Condition for Operation is applicable. Under the terms of this specification, for example, during-initial plant startup or following extended plant outage, the surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment to Operable status.

Bases 3.1:

The reactor protection system automatically initiates a reactor scram to:

1. preserve the integrity of the fuel cladding;
2. preserve the integrity of the primary system barrier; and
3. minimize the energy which must be absorbed, and prevent criticality following a loss of coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function.

The reactor protection system is of the dual channel type. Ref. Section 7.7.1 FSAR. The system is made up of two independent trip systems, each having three subchannels or tripping devices. One of the three subchannels has inputs from the manual scram push buttons and reactor mode switch. Each of the two remaining subchannels has an input from at least one independent sensor which monitors each critical parameter. The outputs of these subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of the proposed IEEE Standard for Nuclear Power Plant Protection Systems issued Sept. 13, 1966. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system. Ref. APED 5179.

The required conditions when the minimum conditions are not met are chosen so as to bring plant operation promptly to such a condition that the particular protection instrument is not required; or the plant is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions.

When the minimum requirements for the number of operable or operating trip systems and instrumentation channels are satisfied, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

Bases 3.1 (Continued):

Three APRM instrument channels are provided for each protection trip system. APRM's #1 and #3 operate contacts in one subchannel, and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5, and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system. Additional IRM channels have also been provided to allow for bypassing of one such channel in each trip system.

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.3 and 2.4.

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by the scram can be accommodated in the discharge piping. Part of this piping consists of two instrument volumes which accommodate in excess of 56 gallons of water each and is the low point in the piping. During normal operation the discharge volumes are empty; however, should they fill with water, the water discharge to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volumes which alarm and scram the reactor when the volume of water in either of the discharge volume receiver tanks reaches 56 gallons. At this point there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of

Bases 3.1 (Continued):

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. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted February 13, 1973. The condenser low vacuum scram is a back-up to the 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.

The main steamline isolation valve closure scram is set to scram when the isolation valves are $\leq 10\%$ 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

Bases 3.1 (Continued):

start-up and intermediate power ranges. Ref. Section 7.4.4 FSAR. A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions. Ref. Section 7.4.3 FSAR. Thus, the IRM is required in the "Refuel" and "Startup" modes. In the power range the APRM system provides required protection. Ref. Section 7.4.5.2 FSAR. Thus, the IRM system is not required in the "Run" mode. The APRM's cover only the power range, the IRM's provide adequate coverage in the start-up and intermediate range, and therefore, the APRM's are not required for the "Refuel" or "Startup" modes.

The high reactor pressure, high drywell pressure, and reactor low water level scrams are required for all modes of plant operation unless the reactor is subcritical and depressurized. They are, therefore, required to be operational for all modes of reactor operation except in the "Refuel" mode with the reactor subcritical and reactor temperature less than 212°F as allowed by Note 3.

The scram discharge volume high level trip function is required for all modes with the exception that it may be bypassed in the "Refuel Mode" under the provisions of Table 3.1.1, allowable by-pass condition (a). In order to reset the safety system after a scram condition, it is necessary to drain the scram discharge volume to clear this scram input condition. This condition usually follows any scram, no matter what the initial cause might have been. Since all of the control rods are completely inserted following a scram it is permissible to bypass this condition because a control rod block prevents withdrawal as long as the switch is in the bypass condition for this function.

To permit plant operation to generate adequate steam and pressure to establish turbine seals and condenser vacuum at relatively low reactor power, the main condenser vacuum trip is bypassed until 600 psig. This bypass also applies to the main steam isolation valves for the same reason. Ref. Section 7.7.1.2 FSAR.

An automatic bypass of the turbine control valve fast closure scram and turbine stop valve closure scram is effective whenever the turbine first stage pressure is less than 30% of its rated value. This insures that reactor thermal power is less than 45% of its rated value. Closure of these valves from such a low initial power level does not constitute a threat to the integrity of any barrier to the release of radioactive material.

Bases 3.1 (Continued):

The IRMs are calibrated by the heat balance method such that 120/125 of full scale on the highest IRM range is below 20% of rated neutron flux (see Specification 2.3.A.2). The requirement that the IRM detectors be inserted in the core assures that the heat balance calibration is not invalidated by the withdrawal of the detector.

Although the operator will set the set points within the trip settings specified on Table 3.1.1, 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, such deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

Trip Function	Deviation	Trip Function	Deviation
3. High Flux IRM	+2/125 of scale	*7. Reactor Low Water Level	-6 inches
5. High Reactor Pressure	+10 psi	8. Scram Discharge Volume High Level	+1 gallon
6. High Drywell Pressure	+1 psi	9. Turbine Condenser Low Vacuum	-1/2 in. Hg

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or 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 the actions specified in 3.1.B are not initiated as specified.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criterion. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable

Bases 3. 1 (Continued):

to permit testing in the other trip system. Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped trip system until the surveillance testing deadline.

The ability to bypass one instrument channel when necessary to complete surveillance testing will preclude continued operation with scram functions which may be either unable to meet the single failure criterion or completely inoperable. It also eliminates the need for an unnecessary shutdown if the remaining channels are found to be operable. The conditions under which the bypass is permitted require an immediate determination that the particular function is operable. However, during the time a bypass is applied, the function will not meet the single failure criterion; therefore, it is prudent to limit the time the bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed.

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Bases 4.1:

The instrumentation in this section will be functionally tested and calibrated at regularly scheduled intervals. Specific surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reactor Protection System," as approved by the NRC and documented in the SER dated July 15, 1987 (letter to T A Pickens from A Thadani).

Calibration frequency of the instrument channel is divided into two groups as defined on Table 4.1.2.

Experience with passive type instruments indicates that a yearly calibration is adequate. Where possible, however, quarterly calibration is performed. For those devices which employ amplifies etc., drift specifications call for drift to be less than 0.5%/month; i.e., in the period of a month a drift of 0.5% would occur and thus provide for adequate margin. For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every three days. Calibration on this frequency assures plant operation at or below thermal limits.

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 other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, 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.

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 3. 2 (Continued):

This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, and RWCU system valves.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel clad temperatures remain less than 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Sections 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "refuel" and "Startup" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 825 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Reference License Amendment Request Dated December 1, 1975 from L. O. Mayer (NSP) to R. S. Boyd (USNRC).

Bases 3.2 (Continued):

The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of HPCI and 300% of RCIC design flows and valve closure times are such that the core will not be uncovered and fission product release will not exceed 10 CFR 100 guidelines.

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

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR remains above the Safety Limit (T.S.2.1.A). The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents operation significantly above the licensing basis power level especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The operator will set the APRM rod block trip settings no greater than that stated in Table 3.2.3. However, the actual setpoint can be as much as 3% greater than that stated in Table 3.2.3 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

The RBM provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. With one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked. Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

Bases 3.2 (Continued):

The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

The IRM rod block function provides local as well as gross core protection The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the Safety Limit (T.S.2.1.A).

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale rod blocks assure that there will be proper overlap between the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation. The downscale trips are set at 3/125 of full scale.

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.

Four radiation monitors (two reactor building vent plenum and two refueling floor) are provided which initiate isolation of the reactor building and operation of the standby gas treatment system following a refueling accident. The monitors measure radioactivity in the reactor building ventilation exhaust and on the refueling floor. One upscale trip signal or two downscale/inoperable trip signals, from a pair of monitors performing the same function, will cause the desired action. Trip settings of 100 mR/hr for the reactor building vent plenum monitors and the refueling floor monitors 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 recirculation pump trip description and performance analysis is discussed in Topical Report NEDO-25016, September 1976, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant". (See September 15, 1976 letter from Mr L O Mayer, NSP, to Mr D L Ziemann, USNRC.) The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction

Bases 3.2 (Continued):

increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

The ATWS high reactor pressure and low-low water level logic also initiates the Alternate Rod Injection System. Two solenoid valves are installed in the scram air header, upstream of the hydraulic control units. Each of the two trip systems energizes a valve to vent the header and causes rod insertion. This greatly reduces the long term consequences of an ATWS event.

Voltage sensing relays are provided on the safeguards bus to transfer the bus to an alternate source when a loss of voltage condition or a degraded voltage condition is sensed. On loss of voltage this transfer occurs immediately. The transfer on degraded voltage has a time delay to prevent transfer during the starting of large loads. The degraded voltage setpoint corresponds to the minimum acceptable safeguards bus voltage for starting and running loads during a loss of coolant accident. An allowance for relay tolerance is included.

Safety/relief valve low-low set logic is provided to prevent any safety/relief valve from opening when there is an elevated water leg in the respective discharge line. A high water leg is formed immediately following valve closure due to the vacuum formed when steam condenses in the line. If the valve reopens before the discharge line vacuum breakers act to return water level to normal, water clearing thrust loads on the discharge line may exceed their design limit. The logic reduces the opening setpoint and increases the blowdown range of three non-APRS valves following a scram. A 15-second interval between subsequent valve actuations is provided assuming one valve fails to

Bases 3.2 (Continued):

open and instrumentation drift has caused the nominal 80-psi blowdown range to be reduced to 60 psi. Maximum water leg clearing time has been calculated to be less than 6 seconds for the Monticello design. Inhibit timers are provided for each valve to prevent the valve from being manually opened less than 10 seconds following valve closure. Valve opening is sensed by pressure switches in the valve discharge line. Each valve is provided with two trip, or actuation, systems. Each system is provided with two channels of instrumentation for each of the above described functions. A two-out-of-two-once logic scheme ensures that no single failure will defeat the low-low set function and no single failure will cause spurious operation of a safety/relief valve. Allowable deviations are provided for each specified instrument setpoint. Setpoints within the specified allowable deviations provide assurance that subsequent safety/relief valve actuations are sufficiently spaced to allow for discharge line water leg clearing.

Control room habitability protection instrumentation assures that the control room operators will be adequately protected against the effects of accidental releases of radioactive leakage which may bypass secondary containment following a loss of coolant accident or radioactive releases from a steam line break accident, thus assuring that the Monticello Nuclear Generating Plant can be operated or shutdown safely.

Although the operator will set the setpoints within the trip settings specified in Tables 3.2.1 through 3.2.9, 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:

Bases 3.2 (Continued):

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Reactor Building Vent Plenum Monitors	+5 mR/hr
	Refueling Floor Radiation Monitors	+5 mR/hr
	* Low Low Reactor Water Level High Drywell Pressure	-3 inches +1 psi
Primary Containment Isolation Functions Table 3.2.1	* Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2%
	High Temp. in Main Steam Line Tunnel	+10°F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	* Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F
	Shutdown Cooling Supply ISO	+7 psi

Bases 3.2 (Continued):

	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
	Reactor Low Pressure (Pump Start) Permissive Bypass Timer	>10 min <24 min
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive)	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale IRM Upscale	-2/125 of Scale +2/125 of Scale
	APRM Downscale APRM Upscale	-2/125 of Scale See Basis 3.2
	RBM Downscale RBM Upscale	-2/125 of Scale +2/125 of Scale
	Scram Discharge Volume-High Level	+ 1 gallon
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure * Low Reactor Water Level	+12 psi -3 Inches
Instrumentation for Safeguards Bus Protection	Degraded Voltage	≥3897 volts (trip) ≤3975 volts (reset) ≥5 sec ≤10 sec (delay)
	Loss of Voltage	<3000 volts >2000 volts

Bases 3.2 (Continued):

	Trip Function	Deviation
Instrumentation for Safety/Relief Valve Low Low Set Logic	Reactor Coolant System Pressure for Opening/Closing	± 20 psig
	Opening- Closing Pressure	≥ 60 psi
	Discharge Pipe Pressure Inhibit	± 10 psid
	Timer Inhibit	-3 sec +10 sec
Other Instrumentation	* High Reactor Water Level	+6 inches
	* Low-Low Reactor Water Level	-3 inches
	Low Condensate Storage Level	-6 inches

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

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.

Bases 4.2:

The instrumentation in this section will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goals are applied. As discussed in Section 4.1 Bases, monthly or quarterly testing is generally specified unless the testing must be conducted during refueling outages. Quarterly calibration is specified unless the calibration must be conducted during refueling outages. Where applicable, sensor checks are specified on a once/shift or one/day basis.

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Bases 3.3/4.3:

A. Reactivity Limitations

1. Reactivity Margin - core loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk at the beginning of the cycle, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in % Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., at the beginning of the cycle. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter. See Figure 3.3.2 of the FSAR for such a curve.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. New values of R must be calculated for each new fuel cycle.

The 0.25% Δk in the expression R + 0.25% Δk is provided as a finite, demonstrable, sub-criticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% Δk in reactivity. Observation of sub-criticality in this condition assures sub-criticality with not only the strongest rod fully withdrawn but at least a R + 0.25% Δk margin beyond this.

2. Reactivity margin - stuck control rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for inoperable control rods, which shall meet this Specification, will be available to the operator. The number of rods permitted to be inoperable could be many more than the six allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than six could be indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housing.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movements after the reactor is critical would indicate an uncoupled condition.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.5.l of the FSAR and the design evaluation is given in

*To disarm the drive electrically, two amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred since it allows continued cooling water flow and minimizes crud accumulation in the drive.

Bases 3.3/4.3 (Continued):

Section 6.5.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta k supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.(3) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. This 0.013 delta k limit, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

Recent improvements in analytical capability have allowed more refined analysis of the control rod drop accident. These techniques have been described in a topical report and two supplements.(1)(2)(3) By using the analytical models described in these reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 10% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

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(1) Paone, C J, Stirn R C and Wooley, J A, "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

(2) Stirn, R C, Paone, C J, and Young, R M, "Rod Drop Accident Analysis for Large BWR's," Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R C, Paone, C J, and Haun, J M, "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

Bases 3.3/4.3 (Continued):

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 delta k limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in reference 4. Each core reload will be analyzed to show conformance to the limiting parameters.

a. A startup inter-assembly local power peaking factor of 1.30 or less. (5)

b. An end of cycle delayed neutron fraction of 0.005.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 delta k. Further, the addition of 0.013 delta k worth of reactivity as a result of a rod drop and in a conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013 delta k limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

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(4) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3.(a)" transmitted by letter from L. O. Mayer (NSP) to J. F. O'Leary (USAEC) dated October 4, 1973.

(5) To include the power spike effect caused by gaps between fuel pellets.

Bases 3.3/4.3 (Continued):

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses twice that previously reported in the FSAR, but still well below the guideline values of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power differences.

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Reference Section 7-9 FSAR. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service when required, a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 10% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup after May 1, 1974.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The

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^{-8} 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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit (T.S.2.1.A). This requires the negative reactivity insertion in any local region of the core and in the overall core to be equivalent to at least the scram reactivity curve used in the transient analysis. 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 at the specified rate locally and in the overall core. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram. 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 MCPR remains above the Safety Limit (T.S.2.1.A).

Bases 3.3/4.3 (Continued):

The analysis assumes 50 milliseconds for Reactor Protection System delay, 200 milliseconds from de-energization of scram solenoids to the beginning of rod motion, and 175 milliseconds later the rods are at the 5% position.

The scram times for all control rods will be determined during each operating cycle. The weekly control rod exercise tests serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram since if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram. Allowing for monthly exercising of one rod in any two by two array is consistent with the bases for local and overall core reactivity insertion rates assumed in the transient analyses discussed above. The frequency of exercising the control rods under the conditions of two or more control rods out of service provides even further assurance of the reliability of the remaining control rods.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds six, the allowable number of inoperable rods.

D. Control Rod Accumulators

The basis for this specification was not described in the FSAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $k_{eff} < 1.0$ -- other repeating rod sequences with more rods withdrawn resulted in $k_{eff} > 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory at any base equilibrium core state to predicted rod inventory at that state. Rod inventory predications can be normalized to actual initial steady state rod patterns to minimize calculational uncertainties. Experience with other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one per cent in reactivity.

<u>Bases 3.3/4.3 (Continued)</u>:

Deviations beyond this magnitude would not be expected and would require thorough evaluations. One per cent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor systems.

As was noted above reactivity anomalies can be found by comparison of the actual control rod inventory to the predicted inventory at a selected base condition. For example, the predicted control rod inventory at 100% power at a specified point in time can be compared to the actual control rod inventory at 100% power and at the specified time to determine if a reactivity anomaly exists. The Monticello Plant has been designed to increase or decrease power level as the system load demand changes. For this type of plant an equilibrium condition of the variables important to making a control rod inventory prediction, specifically the reactivity effects of the xenon, is rarely achieved. The uncertainties of calculating the control rod inventory with non-equilibrium xenon conditions can result in errors which can be misconstrued as reactivity anomalies. Therefore, this specification calls for performing of rod inventory comparisons at a time when xenon will not be a source of error.

F. The safety function of the scram discharge volume vent and drain valves is to limit the loss of reactor coolant leaked past the CRD seals while the scram valves are open. To accomplish this, the vent and drain valves must either be in the closed position or close in a timely manner upon scram initiation. The closure time of 30 seconds is based on a letter dated July 25, 1980 to J G Keppler (Region III) from D E Gilberts (NSP) concerning IE Bulletin No. 80-14. Redundant isolation valves have been provided for each vent and drain line. Closure of one of the valves in each line would be sufficient to maintain the integrity of the scram discharge volume.

G. Whenever a specification (or specifications) can not be met for a particular mode of operation, the reactor would be placed in a mode for which the specification (or specifications) are not required. This requires immediate initiation of a reactor shutdown upon discovery that specifications 3.3A through 3.3D are not met.

Bases 3.4/4.4:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of boron in the reactor core in less than 125 minutes sufficient to bring the reactor from full power to a 3% delta k subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional 25% boron concentration margin for possible imperfect mixing of the chemical solution in the reactor water and dilution from the water in the cooldown circuit.

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak.

The ATWS Rule (10 CFR 50.62) requires the addition of a new design requirement to the generic SLC System design basis. Changes to flow rate, solution concentration or boron enrichment to meet the ATWS Rule do not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution" (natural boron enrichment).

The described minimum system parameters (equivalent to 24 gpm, 10.7% concentration and 55 atom percent Boron-10 enrichment) will ensure an equivalent injection capability that meets the ATWS rule requirement.

Boron enrichment concentration, solution temperature, and volume (including check of tank heater and pipe heat tracing system) are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. A reliability analysis indicates that the plant can be operated safely in this manner for ten days. For additional margin, the allowable out of service time has been reduced to seven days.

The only practical time to test the standby liquid control system is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the replacement charges for the tested system are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

Bases 3.4/4.4 (Continued):

B. The solution saturation temperature varies with the concentration of sodium pentaborate. The solution will be maintained at least 5°F above the saturation temperature within the tank and suction piping to guard against precipitation. The 5°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the control room.

Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Boron Enrichment will not vary unless more Boron is added. No deterioration of the Boron-10 enrichment level should occur during system standby operation. Considering these factors, the test intervals have been established.

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A. ECCS Systems

The core spray system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI mode of the RHR system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the automatic depressurization system (ADS).

The Core Spray System is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining. The Core Spray pump is designed to deliver greater than or equal to 3020 gpm (the SAFER/GESTR-LOCA safety analysis assumed a Core Spray Pump flow of 2,800 gpm, or 2,700 gpm flow into the core + 100 gpm to account for ECCS bypass leakage) against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure.

The surveillance requirements provide adequate assurance that the Core Spray System will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four pumps are available to provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS. LPCI Loop Selection Logic determines which Recirculation loop the four RHR pumps will pump into. Each RHR pump was designed to deliver greater than or equal to 4000 gpm (the safety analysis assumed two pumps delivering 7,740 gpm) against a system head corresponding to a reactor pressure of 20 psi greater than containment pressure.

The allowed out-of-service conditions (Section 3.5.A.3) are determined from ECCS analysis cases analyzed. Only one of these conditions is permitted to exist. If more than one condition exists, an orderly shutdown shall be initiated. A LPCI injection path consists of the two motor operated injection valves on that path.

Bases 3.5/4.5 (Continued):

The surveillance requirements provide adequate assurance that the LPCI system will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which Core Spray system operation or LPCI mode of the RHR system operation maintains core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 3000 gpm (safety analyses assumed 2700 gpm) at reactor pressures between 1120 and 150 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCI system inoperable, adequate core cooling is assured by the operability of the redundant and diversified automatic depressurization system and both the Core Spray and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated operability of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be operable whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

Bases 3.5/4.5 (Continued):

ADS automatically controls three selected safety-relief valves although the safety analysis only takes credit for two valves. It is therefore appropriate to permit one valve to be out-of-service for up to 7 days without materially reducing system reliability.

B. RHR Intertie Line

An intertie line is provided to connect the RHR suction line with the two RHR loop return lines. This four-inch line is equipped with three isolation valves. The purpose of this line is to reduce the potential for water hammer in the recirculation and RHR system when required to cooldown with an isolated or idle recirculation system. The isolation valves are opened during a cooldown to ensure a uniform cooldown of the RHR injection piping. If one recirculation loop is isolated or idle, these valves and associated piping allow the operating loop to cool the isolated or idle loop. The RHR loop return line isolation valves receive a closure signal on LPCI initiation. in the event of an inoperable return line isolation valve, there is a potential for some of the LPCI flow to be diverted to the broken loop during a loss of coolant accident. Surveillance requirements have been established to periodically cycle the RHR intertie line isolation valves. In the event of an inoperable RHR loop return line isolation valve, either the inoperable valve is closed or the other two isolation valves are closed to prevent diversion of LPCI flow. The RHR intertie line flow is not permitted in the Run Mode to eliminate 1) the need to compensate for the small change in jet pump drive flow or 2) a reduction in core flow during a loss of coolant accident.

C. Containment Spray/Cooling Systems

Two containment spray/cooling subsystems of the RHR system are provided to remove heat energy from the containment and control torus and drywell pressure in the event of a loss of coolant accident. A containment spray/cooling subsystem consists of 2 RHR service water pumps, a RHR heat exchanger, 2 RHR pumps, and valves and piping necessary for Torus Cooling and Drywell Spray. Torus Spray is not considered part of a containment spray/cooling subsystem. Placing a containment spray/cooling subsystem into operation following a loss of coolant accident is a manual operation.

The most degraded condition for long term containment heat removal following the design basis loss of coolant accident results from the loss of one diesel generator. Under these conditions, only one RHR pump and one RHR service water pump in the redundant division can be used for containment spray/cooling. The containment temperature and pressure have been analyzed under these conditions assuming service water and initial suppression pool temperature are both 90°F. Acceptable margins to containment design conditions have been demonstrated. Therefore the containment spray/cooling system is more than ample to provide the required heat removal capability. Refer to USAR Sections 5.2.3.3, 6.2.3.2.3, and 8.4.1.3.

During normal plant operation, the containment spray/cooling system provides cooling of the suppression pool water to maintain temperature within the limits specified in Specification 3.7.A.1.

Bases 3.5/4.5 (Continued):

The surveillance requirements provide adequate assurance that the containment spray/cooling system will be operable when required. The head and flow requirements specified for the RHR service water pumps provide assurance that the minimum required service water flow can be supplied to the RHR heat exchangers for the most degraded condition for long-term containment heat removal following the design basis loss of coolant accident.

D. 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.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

F. Recirculation System

The reactor is designed such that thermal hydraulic oscillations are prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal-hydraulic instability, a power-flow exclusion region, to be avoided during normal operation, is calculated using the approved methodology as stated in specification 6.7.A.7. Since the exclusion region may change each fuel cycle the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in this region and to immediately exit upon an entry. Entries into the exclusion region are not part of normal operation. An entry may occur as the result of an abnormal event such as a single recirculation pump trip. In these events, operation in the exclusion region may be needed to prevent equipment damage, but actual time spent inside the exclusion region is minimized. Though operator action can prevent the occurrence and protect the reactor from an instability, the APRM flow biased scram function will suppress oscillations prior to exceeding the fuel safety limit.

Power distribution controls are established to ensure the reactor is operated within the bounds of the stability analysis. With these controls in place, there is confidence that an oscillation will not occur outside of the stability exclusion region. Without these controls, it is theoretically possible to operate the reactor in such a manner as to cause an oscillation outside of the exclusion region. A nominal 5% power-flow buffer region outside of the exclusion region is provided to establish a stability margin to the analytically defined exclusion region. The buffer region may be entered only when the power distribution controls are in place.

Continuous operation with one recirculation locp was analyzed and the adjustments specified in specification 3.5.F.2 were determined by NEDO-24271, June 1980, "Monticello Nuclear Generating Plant Single Loop Operation". Specification 3.6.A.2 governs the restart of the pump in an idle recirculation loop. Adherence to this specification limits the probability of excessive flux transients and/or thermal stresses.

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Bases 3.6/4.6:

A. Reactor Coolant Heatup and Cooldown

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code.

During reactor operation, the temperature of the coolant in an idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the pump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel thermal and pressure transients.

During hydrostatic pressure testing, a coolant heatup or cooldown of 20°F in any one-hour period has a negligible effect on the reactor operating limits of Figure 3.6.2.

B. Reactor Vessel Temperature and Pressure

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown and during inservice hydrostatic testing were established using 10 CFR 50, Appendix G, May 1983 and Appendix G of the Summer 1976 or later Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These operating limits assure that a large postulated surface flaw, having a depth of 0.24 inches at the flange-to-vessel junction and one-quarter of the material thickness, at all other reactor vessel locations and discontinuity regions can be safely accommodated. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Summer 1966 Addenda).

A General Electric Company procedure, designed to evaluate fracture toughness requirements for older plants where information may be incomplete, was used to estimate RT_{NDT} values on an equivalent basis to the new requirements for plants which have construction permits after August 15, 1973.

Bases 3.6/4.6 (Continued):

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the reactor pressure vessel. Two types of information are needed in this analysis: 1) A relationship between the changes in fracture toughness of the reactor pressure vessel steel and the neutron fluence (integrated neutron flux), and 2) A measure of the neutron fluence at the point of interest in the reactor pressure vessel wall.

The relationship of predicted adjustment of reference temperature versus fluence and the copper and nickel content of the core beltline materials given in Regulatory Guide 1.99, Revision 2, was used to define the core beltline temperature adjustment versus fluence shown on Figure 3.6.1.

A relationship between full power years of operation and neutron fluence has been experimentally determined for the reactor vessel. The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown following nuclear shutdown), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which were installed adjacent to the vessel wall were removed to experimentally determine the neutron fluence versus full power years of operation. This experimental result was updated by testing additional dosimetry removed with the first surveillance capsule.

Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. An analysis and report will be submitted to the Commission on all such surveillance specimens removed from the reactor vessel in accordance with 10 CFR 50, Appendix H, including information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.

Bases 3.6/4.6 (Continued):

The requirements for cold bolt-up of the reactor vessel closure are based on the NDT temperature plus 60°F which is derived from the requirements of the ASME Boiler and Pressure Vessel Code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head and shell material, and stud material is a maximum of 10°F. The minimum temperature for bolt-up is therefore 10° + 60° = 70°F. The neutron radiation fluence at the closure flanges is well below 10^{17} n/cm^2 (E>1 MEV) and therefore radiation effects will be minor and will not influence this temperature.

Bases 3.6/4.6 (Continued):

C. Coolant Chemistry

A steady state radioiodine concentration limit of 5 μCi of I-131 dose equivalent per gram of water in the reactor coolant system can be reached if the gross radioactivity in the gaseous effluents are near their limits or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the nearest site boundary (465 m) to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 5 μCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 30 meters under fumigation conditions for Pasquill type F 1 meter/sec wind speed and a steam line isolation valve closure time of five seconds with a steam/water mass release of 36,000 pounds.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.C.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

Bases 3.6/4.6 (Continued):

When conductivity is in its proper normal range (approximately 10 μmho/cm during reactor startup and 5 μmho/cm during power operation), pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt, e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 5 μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time when the conductivity exceeds 5μmho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.C.1(b) may be performed by a gamma scan and gross beta and alpha determination.

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

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% high. As discussed in the Section 2.2 Bases, the 1375 psig Code limit is not exceeded in any case.

Bases 3.6/4.6 (Continued):

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation). The Low-Low Set and ADS functions are discussed further in Sections 3.2 and 3.5.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety 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.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

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G. Jet Pumps

By monitoring jet pump performance on a prescribed schedule, significant degradation in performance that would precede jet pump failure can be detected. An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it may present a hazard in the event of a large break accident by reducing the capability of reflooding the core; thus, the requirement for shutdown of the reactor with an inoperable jet pump.

The jet pump performance monitoring procedures are comprised of the following tests:

1. Core Flow versus Square Root of Core Plate Differential Pressure: change in core resistance is the main contributor to recirculation system performance changes. If core resistance increases, it requires more energy (pump speed) to produce rated core flow. If resistance decreases, less speed is needed.

2. Recirculation Pump Flow/Speed Ratio: the pump operating characteristic is determined by the flow resistance from the loop suction through the jet pump nozzle. Since this resistance is essentially independent of core power, the flow is linearly proportional to pump speed, making their ratio a constant (flow/RPM is constant). A decrease in the ratio indicates a plug, flow restriction, or loss in pump hydraulic performance. An increase indicates a leak or new flow path between the recirculation pump discharge and jet pump nozzle.

3. Jet Pump Loop Flow/Recirculation Pump Speed Ratio: this relationship is an indication of overall system performance.

4. Jet Pump Differential Pressure Relationships: if a potential problem is indicated, the individual jet pump differential pressures are used to determine if a problem exists since this is the most sensitive indicator of significant jet pump performance degradation.

The data base used to determine the normal operating range for (2) and (3) above is verified during the startup following each refueling outage. Surveillance tests are performed as soon as practical after reaching a pump speed of 60%.

H. Snubbers

All snubbers are required to be operable above Cold Shutdown to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval is determined by the sample population size and the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design feature directly related to rejection of the snubber.

When a snubber is found inoperable, an engineering evaluation or inspection is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The evaluation or inspection will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

Bases 3.6/4.6 (Continued):

To provide assurance of snubber functional reliability, a representative sample of 10% of the installed snubbers will be functionally tested during plant shutdowns at intervals of no more than 18 months ± 25%. Observed failures of these sample snubbers will require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

Bases 3.7:

A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the offsite doses to values less than 10 CFR 100 guideline values in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit incremental control worth to less than 1.3% delta k. A drop of a 1.3% delta k increment of a rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100 guide line values.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the maximum allowable primary containment pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. See USAR Section 5.2.3.2.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 42 psig which is below the allowable pressure of 62 psig.

Bases 3.7 (Continued):

Vent system downcomer submergence is three feet below the minimum specified suppression pool water level. This length has been shown to result in reduced postulated accident loading of the torus while at the same time assuring the downcomers remain submerged under all seismic and accident conditions and possess adequate condensation effectiveness.[3]

The maximum temperature at the end of blowdown tested during the Humboldt Bay[1] and Bodega Bay[2] tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

For an initial maximum suppression chamber water temperature of 90°F and conditions which lead to minimum containment pressure, adequate net positive suction head (NPSH) is maintained for the core spray, RHR, and HPCI pumps under loss of coolant accident conditions. Analyses were performed for a broad range of pump combinations and failure modes to define the minimum amount of containment pressure available to provide adequate NPSH in the short and long term. Refer to Section 5.2.3.3 of the USAR for a discussion of these analyses and figures which demonstrate graphically the amount of pressure required and the minimum containment pressure available to supply the required NPSH for the emergency core cooling pumps in the limiting pump combinations evaluated. No pump cavitation will occur over either the short or long term periods under conditions resulting in minimum containment pressure.

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(1) Robbins, C.H. "Tests of Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) General Electric NEDE-21885-P, "Mark I Containment Program Downcomer Reduced Submergence Functional Assessment Report," June, 1978.

Bases 3.7 (Continued):

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor water temperatures above 212°F provides additional margin above that available at 330°F.

The large amount of water that must be added or removed to cause a significant change in the suppression chamber water inventory is not likely to go un-noticed. With a daily check of water volume, there is an extremely low probability that a loss of coolant accident will occur simultaneously with water volume being outside of the specified range. Two indicators provide redundant readings for comparison (with no automatic action initiation). The provisions allowing one or both indicators out of service are consistent with the need for a redundant indicator and the frequency for checking the volume, respectively.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping.

Bases 3.7 (Continued):

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and between the suppression chamber and reactor building during a loss of coolant accident so that structural integrity of the containment is maintained.

The vacuum relief system between the pressure suppression chamber and reactor building consist of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 1 psi. The external design pressure is 2 psig. One valve may be out of service for repairs for a period of seven days. This period is based on the low probability that system redundancy would be required during this time. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the drywell vacuum relief valves is sufficient to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to less than the design limit of 2 psi. Capacity of the vacuum relief valves has been confirmed using a sizing model developed in conjunction with the Mark I Containment Long Term Program.(2) With six of the eight valves operable, the pressure differential is limited to less than 2 psi and containment integrity is assured.

In addition to the above considerations, postulated leakage through the vacuum breaker to the suppression chamber air space could result in a partial bypass of pressure suppression in the event of a LOCA or a small or intermediate steam leak. This effect could potentially result in exceeding containment design pressure. As a result of the leakage potential, the containment response has been analyzed for a number of postulated conditions. It was found that the maximum allowable bypass area for any postulated break size was equivalent to a six-inch diameter opening.(1) This bypass corresponds to a

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(1) Report on Torus to Drywell Vacuum Breaker Tests and Modifications for Monticello Nuclear Generating Plant, dated March 12, 1973, submitted to Mr D J Skovholt, AEC-DL, from Mr L O Mayer, NSP.

(2) "Monticello Torus-to-Drywell Vacuum Breaker Requirements," Nutech, Inc, December, 1980, submitted as Exhibit C, Northern States Power Company License Amendment Request dated June 4, 1981.

one-inch opening of any one valve or a 1/8-inch opening for all eight valves, measured at the bottom of the disc with the top of the disc at the seat. The position indication system is designed to detect closure within 1/8 inch at the bottom of the disc.

At each refueling outage and following any significant maintenance on the vacuum breaker valves, positive seating of the vacuum breakers will be verified by leak test. The leak test is conservatively designed to demonstrate that leakage is less than that equivalent to leakage through a one-inch orifice which is about 3% of the maximum allowable. This test is planned to establish a baseline for valve performance at the start of each operating cycle and to ensure that vacuum breakers are maintained as nearly as possible to their design condition. This test is not planned to serve as a limiting condition for operation.

During reactor operation, an exercise test of the vacuum breakers will be conducted monthly. This test will verify that disc travel is unobstructed and will provide verification that the valves are closing fully through the position indication system. If one or more of the vacuum breakers do not seat fully as determined from the indicating system, a leak test will be conducted to verify that leakage is within the maximum allowable. Since the extreme lower limit of switch detection capability is approximately 1/16", the planned test is designed to strike a balance between the detection switch capability to verify closure and the maximum allowable leak rate. A special test was performed to establish the basis for this limiting condition. During the first refueling outage all ten vacuum breakers were shimmed 1/16" open at the bottom of the disc. The bypass area associated with the shimming corresponded to 63% of the maximum allowable.[1] The results of this test are shown in Figure 3.7.1. Two of the original ten vacuum breakers have since been removed.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening closing cycle, the position indicating lights at the remote test panels are designed to function as follows:

Fully Closed	2 Green	-	On
	2 Red	-	Off
Intermediate Position	2 Green	-	Off
	2 Red	-	Off
Fully Open	2 Green	-	Off
	2 Red	-	On

The remote test panels consist of indication and controls in the control room and indication in the reactor building. The control room indication and controls for the drywell to suppression chamber vacuum breakers consist of one red light and one green light for each of the eight valves, a common vacuum breaker selector switch, and a common test switch. The reactor building vacuum breaker panel contains one red light and one green light for each of the eight valves. There are four independent limit switches on each valve. The two switches controlling the green lights are adjusted to provide an indication of disc opening of less than 1/8" at the bottom of the disc. These switches are also used to activate the valve position alarm circuits. The two switches controlling the red lights are adjusted to provide indication of the disc very near the full open position.

The control room alarm circuits are redundant and fail safe. This assures that no simple failure will defeat alarming to the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or to investigate possible changes in valve position status, or both. If the alarm cannot be cleared due to the inability to establish indication of closure of one or more valves, additional testing is required. The alarm system allows the operator to make this evaluation on a timely basis. The frequency of the testing of the alarms is the same as that required for the position indication system.

Operability of a vacuum breaker valve and the four associated indicating light circuits shall be established by cycling the valve. The sequence of the indicating lights will be observed to be that previously described. If both green light circuits are inoperable, the valve shall be considered inoperable and a pressure test is required immediately and upon indication of subsequent operation. If both red light circuits are inoperable, the valve shall be considered inoperable, however, no pressure test is required if positive closure indication is present.

Oxygen concentration is limited to 4% by volume to minimize the possibility of hydrogen combustion following a loss of coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems failed to sufficiently cool the core. The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week the oxygen concentration will be determined as added assurance.

B. Standby Gas Treatment System and C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading prior to initial power testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment system circuit is designed to automatically start upon containment isolation and to maintain the reactor building pressure at the design negative pressure so that all leakage should be in-leakage. Should one circuit fail to start, the redundant alternate standby gas treatment circuit is designed to start automatically. Each of the two circuits has 100% capacity. Only one of the two standby gas treatment system circuits is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

Bases 3.7 (Continued):

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the Primary Containment. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the Primary Containment isolation valves are discussed in Section 5.2 of the USAR. A listing of all Primary Containment automatic isolation valves including maximum operating time is given in USAR Table 5.2-3b.

E. Combustible Gas Control System

The function of the Combustible Gas Control System (CGCS) is to maintain oxygen concentrations in the post-accident containment atmosphere below combustible concentrations. Oxygen may be generated in the hours following a loss of coolant accident from radiolysis of reactor coolant.

The Technical Specifications limit oxygen concentrations during operation to less than four percent by volume during operation. The maintenance of an inert atmosphere during operation precludes the build-up of a combustible mixture due to a fuel metal-water reaction. The other potential mechanism for generation of combustible mixtures is radiolysis of coolant which has been found to be small.

A special report is required to be submitted to the Commission to outline CGCS equipment failures and corrective actions to be taken if inoperability of one train exceeds thirty days. In addition, if both trains are inoperable for more than 30 days, the plant is required to shutdown until repairs can be made.

Bases 4.7:

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident. Daily checks are specified of pool temperature and volume to ensure that these parameters are within their allowable ranges.

The interiors of the drywell and suppression chamber are painted to prevent corrosion. The inspection of the paint during each refueling outage, approximately once per year, assures the paint is intact and is not deteriorating. Experience with this type of paint indicates that the inspection interval specified is adequate.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The design basis loss of coolant accident was analyzed at the primary containment maximum allowable accident leak rate of 1.2% and has been evaluated by the NRC Staff(1). Computed offsite doses are well below the guidelines of 10 CFR Part 100.

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(1) Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, in the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970, Section 4.1.

Bases 4.7 (Continued):

While the design of the Monticello plant predates 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," testing substantially conforms to the requirements of Appendix J. The design of the plant was thoroughly reviewed to determine where compliance with Appendix J was impossible or impractical. In each case where a departure from the requirements of Appendix J was identified, a request for exemption from the requirements of Appendix J or a plant modification was proposed and submitted for NRC Staff review. Exemptions were proposed in those cases where compliance with Appendix J would have provided no meaningful improvement in plant safety.

In their review of Appendix J compliance[1], the NRC Staff approved a number of exemption requests, denied others, and provided necessary interpretation and clarification of the requirements of Appendix J. The Technical Specification surveillance requirements reflect the results of this review.

Exemption from the requirements of Appendix J was provided in the following areas:

a. Testing of valves sealed by water

b. Low pressure testing of main steam line isolation valves.

c. Low pressure testing of the primary containment airlock

d. Reduced airlock testing frequency when the airlock is in frequent use

The Monticello airlock is tested by pressurizing the space between the inner and outer doors. Individual door seal leakage tests cannot be performed. Since the inner door is designed to seat with containment pressure forcing the door closed, special bracing must be installed for each leakage test. The outer door must be opened to install and remove this bracing. Because of the complexity of this operation, up to 24 hours may be necessary to perform a leakage test.

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(1) Letter from D G Eisenhut, Director, Division of Licensing, USNRC, dated June 3, 1984, "Safety Evaluation by the Office of NRR, Appendix J Review".

Bases 4.7 (Continued):

On September 26, 1995, Regulatory Guide 1.163 became effective providing guidance on performance based testing to the requirements of 10 CFR 50, Appendix J, Option B. Monticello has adopted Option B, Section III.A of 10 CFR Part 50, Appendix J, for Type A primary reactor containment integrated leakage rate testing. Monticello will continue to perform Type B and C testing in accordance with 10 CFR Part 50, Appendix J, Option A.

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B. Standby Gas Treatment System, and C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability. Secondary Containment Capability Test data obtained under non-calm conditions is to be extrapolated to calm wind conditions using information provided in "Summary Technical Report to the United States Atomic Energy Commission, Directorate of Licensing, on Secondary Containment Leak Rate Test", submitted by letter dated July 23, 1973, and as described in NSP letter to the NRC dated August 18, 1995, with subject, "Revision 2 to License Amendment Request Dated June 8, 1994, Standby Gas Treatment and Secondary Containment Technical Specifications."

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system inplace testing procedures will be established utilizing applicable sections of ANSI N510-1989 standard as a procedural guideline only. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 Revision 2 (March 1978). The charcoal adsorber efficiency test procedures will allow for the removal of a representative sample. The 30°C, 95% relative humidity test per ASTM D 3803-89 is the test method to establish the methyl iodine removal efficiency of adsorbent. The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. 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 inplace testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52 Revision 2 (March 1978). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inline heaters at rated power, automatic initiation of each standby gas treatment system circuit, and leakage tests after maintenance or testing which could affect leakage, is necessary to assure system performance capability.

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 in USAR Table 5.2-3b 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.

The primary containment isolation valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensor and associated trip channels are also checked to demonstrate the capability for automatic isolation. Reference Section 5.2.2.5.3 and Table 5-2-3b USAR. 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.

Bases 4.7 (Continued):

The containment is penetrated by a large number of small diameter instrument lines. A program for the periodic testing (see Specification 4.7.D) and examination of the valves in these lines has been developed and a report covering this program was submitted to the AEC on July 27, 1983.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

E. Combustible Gas Control System

The Combustible Gas Control System (CGCS) is functionally tested once every six months to ensure that the recombiner trains will be available if required. In addition, calibration and maintenance of essential components is specified once each operating cycle.

Bases 3.8/4.8:

A. Liquid Effluents

Specification 3.3.A.1 is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in the Mississippi River will not result in exposures exceeding (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

Specification 3.8.A.2.a is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. Action required by Specification 3.8.A.2.b provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". Considering that the nearest drinking water supply using the receiving water is 33 river miles downstream, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, Revision 1," April 1977. NUREG-0133, October, 1978, provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

Bases 3.8/4.8 (Continued):

Specification 3.8.A.3 provides assurance that the liquid radwaste treatment system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

Restrictions on the quantity of radioactive liquid material contained in tanks are required only for temporary tanks. All exterior permanent tanks are diked to prevent release of their contents in the event of leakage. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

Surveillance requirements for continuous liquid release points are not provided since all Monticello releases are "batch" type releases.

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 setpoints for these instruments will be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. The operability requirements for instrumentation are consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Bases 3.8/4.8 (Continued):

B. Gaseous Effluents

Specification 3.8.B.1.a is provided to ensure that the dose rate at anytime at the site boundary from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the restricted area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the restricted area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the restricted area boundary to 500 mrem/year to the total body or to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to $\leq$1500 mrem/year at the site boundary.

Specification 3.8.B.2.a is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation implement the guides set forth in Section II.B of Appendix I. Action required by Specification 3.8.B.2.b provides the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the restricted area boundary may be based upon the historical average atmospheric conditions. NUREG-0133, October, 1978 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

Bases 3.8/4.8 (Continued):

Specification 3.8.B.3.a is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The release rate specifications for I-131, tritium, and radioactive particulates with half-lives greater than eight days are dependent on the existing radionuclide pathways to man in the unrestricted area. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

Specification 3.8.B.4 provides assurance that the offgas treatment system will be in operation whenever main condenser offgas is released to the environment. The requirement that the appropriate portions of this system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

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 will be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability requirements for this instrumentation are consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Bases 3.8/4.8 (Continued):

Specification 3.8.B.4.c is provided to ensure that the concentration of potentially explosive gas mixtures contained in the compressed storage subsystem is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. Maintaining the concentration of hydrogen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.4.e is provided to limit the radioactivity which can be stored in one decay tank. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the site restricted area boundary will not exceed 20 mrem. A flow restrictor in the discharge line of the decay tanks prevents a tank from being discharged at an uncontrolled rate. In addition, interlocks prevent the contents of a tank from being released with less than 12 hours of holdup.

Specification 3.8.B.5 establishes a maximum activity at the steam jet air ejector. Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the restricted area boundary will not exceed the limits of 10 CFR Part 20 in the event this effluent is inadvertently discharged directly to the environment with minimal treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.6 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. During outages when the containment is opened for maintenance, the containment ventilation exhaust is directed to the monitored reactor building vent. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the control room operators to monitor the activity level of the resulting effluent by use of the Reactor Building Vent Wide Range Gas Monitors. In the event that the reactor building release rate exceeds the Reactor Building Vent Wide Range Gas Monitor alarm settings, the monitors will alarm in the control room alerting the operators to take actions to limit the release of gaseous radioactive effluents. The alarm settings for the reactor building vent wide range gas monitors are calculated in accordance with the NRC approved methods in the ODCM to ensure that alarms will alert control room operators prior to the limits of specification 3.8.B.1 being exceeded.

C. Solid Radioactive Waste

Specification 3.8.C.1 provides assurance that the solid radwaste system will be used whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

D. Dose from All Uranium Fuel Cycle Sources

Specification 3.8.D is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a special report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. Submittal of the report is considered a timely request and a variance is granted until Staff action on the request is complete. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a real individual will exceed 40 CFR 190 if the individual reactors remain with the reporting requirement level. For the purpose of the special report it may be assumed that the dose commitment to the real individual from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

Bases 3.9:

The general objective is to assure an adequate supply of power with at least one active and one standby source of power available for operation of equipment required for a safe plant shutdown, to maintain the plant in a safe shutdown condition, and to operate the required engineered safeguards equipment following an accident.

AC for shutdown requirements and operation of engineered safeguards equipment can be provided by either of the two standby sources of power (the diesel generators) or any of the three active sources of power (No. 1R, No. 2R, or No. 1AR transformers). Refer to Section 8 of the USAR.

To provide for maintenance and repair of equipment and still have redundancy of power sources, the requirement of one active and one standby source of power was established. The plant's main generator is not given credit as a source since it is not available during shutdown.

The plant 250 V dc power is supplied by two batteries. Most station 250 V loads are supplied by the original station 250 V battery. A new 250 V battery has been installed for HPCI loads and may be used for other station loads in the future. Each battery is maintained fully charged by two associated chargers which also supply the normal dc requirements with the batteries as a standby source during emergency conditions. The plant 125 V dc power is normally supplied by two batteries, each with an associated charger. Backup chargers are available.

The minimum diesel fuel supply of 34,500 gallons will supply one diesel generator for a minimum of seven days of full load (2500 KW) operation. Actual fuel consumption during this period would be 33,096 gallons, but the minimum tank level has been established at the higher 34,500 gallon value to allow for instrument inaccuracy, tank volume uncertainties, and the location of the suction piping within the tank. Additional diesel fuel can normally be obtained within a few hours. Maintaining at least 7 days supply is therefore conservative.

In the normal mode of operation, power is available from the offsite sources. One diesel may be allowed out of service based on the availability of offsite power provided that the remaining diesel generator is demonstrated to be operable within 24 hours. This test is required even if the inoperable diesel is restored to operability within 24 hours. Thus, though one diesel generator is temporarily out of service, the offsite sources are available, as well as the remaining diesel generator. Based on a monthly testing period (Specification 4.9), the seven day repair period is justified. (1)

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(1) "Reliability of Engineered Safety Features as a Function of Testing Frequency", I M. Jacobs, Nuclear Safety, Volume 9, No. 4, July - August 1968.

Bases 4.9:

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel will be manually started, synchronized to the bus and load picked up. It is expected that the diesel generator will be run for one to two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required. In addition, during the test when the generator is synchronized to the bus it is also synchronized to the offsite power source and thus not completely independent of this source. To maintain the maximum amount of independence, a thirty day testing interval is also desirable.

Each diesel generator has two air compressors and six air receiver tanks for starting. It is expected that the air compressors will run infrequently. During the monthly check of the diesel, the receivers will be drawn down below the point at which the compressor automatically starts to check operation and the ability of the compressors to recharge the receivers. Pressure indicators are provided on each bank of three receivers. During the monthly load test of the diesel generators, the diesel fuel oil transfer pump and diesel oil service pump will be operated. A sample of diesel fuel will be taken monthly to assure that the quality remains high.

The test of the emergency diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system, i.e., it will check diesel starting and closure of diesel breaker and sequencing of loads on the diesel. The diesel will be started by simulation of a loss of coolant accident. In addition, an undervoltage condition will be imposed to simulate a loss of offsite power. The timing sequence will be checked to assure proper loading in the time required. The only load on the diesel is that due to friction and windage and a small amount of bypass flow on each pump. Periodic tests between refueling outage check the diesel to run at full load and the pumps to deliver full flow. Periodic testing of the various components plus a functional test at a refueling interval are sufficient to maintain adequate reliability.

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

In addition, the checks described also provide adequate indication that the batteries have the specified ampere-hour capability.

Bases 3.10/4.10:

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

For a new core the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's, one in and one adjacent to any core quadrant where fuel or control rods are being moved, assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the SRM. In this case only, the SRM or dunking type detector count rate is permitted to be less than 3 counts per second.

C. Fuel Storage Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37'9") and well above a level to assure adequate cooling.

D. Minimum Shutdown Period

A minimum shutdown period of 24 hours is specified prior to movement of fuel within the reactor since analysis of refueling accidents assume a 24-hour decay time following extended operation at power. Since the reactor must be shut down, depressurized, and the head removed prior to moving fuel, it is not expected that fuel could actually be moved in less than 24 hours.

<u>Bases 3.10/4.10 (Continued)</u>:

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as removal of temporary control curtains, control rod drive maintenance, in-service inspection requirements, examination of the core support plate, etc. When the refueling interlock input signal from a withdrawn control rod is bypassed, administrative controls will be in effect to prohibit fuel from being loaded into that control cell.

These operations are performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in Part A of these Bases. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with the control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

Bases 3.11:

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design bases loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than $\pm 20°$ relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures at rated conditions conform to 10 CFR 50.46. The limiting value for APLHGR is given by this specification.

The flow dependent correction factor provided in the Core Operating Limits Report applied to the rated condition's APLHGR limits assures that 1) the 2200°F PCT limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and 2) the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from less than rated core flow conditions. The power dependent correction factor provided in the Core Operating Limits Report applied to the rated conditions APLHGR limits assures that the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from all conditions provided in the Core Operating Limits Report.

Those abnormal operational transients, analyzed in FSAR Section 14.5 which result in an automatic reactor scram are not considered a violation of LCO. Exceeding APHLGR limits in such cases need not be reported.

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of LCO. Exceeding LHGR limits in such cases need not be reported.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in References 1 and 2 assumed the steady state MCPR prior to the postulated loss-of-coolant accident for all fuel types for rated flow. The Rated

Bases 3.11 (Continued):

MCPR Limit is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

Flow runout events are analyzed with the purpose of establishing a flow dependent MCPR limit that would prevent the Safety Limit CPR from being reached during a flow runout. A flow runout event is a slow flow and power increase which is not terminated by a scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Initial conditions for the transient are set such that the limiting CPR is near the Safety Limit. MCPR values are determined from the resulting change in CPR when core flow is increased to a possible maximum. Several combinations of initial power, flow, and exposure are analyzed to cover the range of operability defined by the power/flow map. The calculated flow dependent MCPR limit ($MCPR_f$) for a given core flow is provided in the Core Operating Limits Report.

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, provided in the Core Operating Limits Report. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are provided in the Core Operating Limits Report. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error.

Bases 3.11 (Continued):

This limit was determined based upon bounding analyses for the limiting transient at the given core power level.

At thermal power levels less than or equal to 25% of rated thermal power, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. MCPR evaluation below this power level is therefore unnecessary. The daily requirement for calculating MCPR above 25% of rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K", NEDE-20566, November, 1975.

2. "Loss-of-Coolant Accident Analysis Report for the Monticello Nuclear Generating Plant", NEDO-24050-1, December, 1980, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (USNRC), February 6, 1981.

Bases 4.11

The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement have caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. In addition, the MCPR is checked whenever changes in the core power level or distribution are made which have the potential of bringing the fuel rods to their thermal-hydraulic limits.

Bases 3.12/4.12

The program, facilities, personnel, and procedures for safe storage, handling, and use of sealed sources containing radioactive materials is described in Supplement No. 2 to the Application for Conversion of DPR-22 to Full Term, submitted by Northern States Power Company on August 16, 1974. The surveillance program described in these specifications is a part of the program to detect and control contamination of areas in the plant by such radioactive materials.

Small quantities of byproduct materials are exempt from licensing by 10 CFR 30.18 and therefore are exempt from leakage tests in these specifications. Inhalation or ingestion of such small quantities of byproduct materials from a sealed source would result in less than one maximum permissible body burden for total body irradiation. Sources containing less than 0.1 microcurie of plutonium are exempt from leakage tests by 10 CFR 70.39(c) and therefore such quantities of special nuclear materials (including alpha emitters) are exempt from leakage tests in these specifications. The acceptance criteria of less than 0.005 microcurie on the test sample is also based on 10 CFR 70.39(c).

Bases 3.13:

Elements of the fire detection and protection system are required to be operable to protect safety related structures, systems, and components whenever those structures, systems, or components are required to be operable. Fire detection and protection systems will normally be maintained operable at all times except for periods of maintenance and testing.

Fire detection instrumentation is installed throughout the plant to protect safety related structures, systems, and components. The detectors in each area initiate a local alarm and an alarm in the control room. All circuits are supervised and the installation meets the requirements of NFPA-72D. The Specifications require all detectors to be operable in those zones having only one detector (battery rooms). In other plant areas, Table 3.13.1 permits one detector in each zone to be inoperable. If more detectors are inoperable, a patrolling fire watch is established in the affected area until the required number of detectors are restored to operable status. The loss of one detector does not significantly degrade the ability to detect fires in areas of the plant having multiple detectors.

The fire suppression water system is supplied by three identical vertical centrifugal pumps rated at 1500 gpm at 100 psig each. Two of these pumps are motor driven and one is diesel driven. One of the motor driven pumps normally supplies the needs of the screen wash system and is designated the screen wash/fire pump. Transfer from screen wash duty to fire duty occurs automatically. All pumps are started automatically by instrumentation sensing header pressure. Any two pumps are capable of supplying all fire fighting water requirements in safety related areas of the plant. If a pump is inoperable, it must be repaired within seven days or a report is submitted to the Commission. If two pumps are inoperable, or if other circumstances interrupt the supply of water to any safety related area, a backup source of water must be provided within 24 hours and the Commission notified.

Automatic sprinkler systems are installed in both diesel-generator rooms and both day tank rooms. Other sprinkler and deluge systems are installed in turbine lube oil piping and storage areas and other non-safety related portions of the plant. An automatic Halon suppression system is installed in the cable spreading room. Inoperability of any of the automatic suppression systems in safety related areas of the plant requires the stationing of a continuous fire watch in the area equipped with backup manual fire suppression equipment. Hose stations and yard hydrant hose houses are provided in all safety related areas of the plant and surrounding all principal plant buildings. These stations are supplied from the fire suppression water system. If the water supply to these areas is interrupted, a hose supplied from an operable source is made available to protect the area having the inoperable station.

Piping and electrical penetrations are provided with seals where required by the fire severity. If a seal is made or found to be inoperable for any reason and equipment protected by that fire barrier is required to be operable, the penetration area is continuously attended until an effective fire seal is restored or the detection system on one side of the barrier is determined to be operable and an hourly fire patrol is established. Seals have been qualified for the maximum fire severity present on either side of the barrier.

Bases 3.13 (Continued):

The alternate shutdown system panel is provided to assure the capability of achieving cold shutdown, external to the control room, in the unlikely event the control room becomes uninhabitable or safe shutdown equipment in the control room or cable spreading room is damaged by fire. Control of those systems on the alternate shutdown system panel is taken when the locking master transfer switch is moved from the normal to the transfer position and each system's individual transfer switch is put in the transfer mode. When control is established at the alternate shutdown system panel no control of those systems is available from the control room and all automatic initiation signals have been disabled. The master transfer switch shall remain in the locked position at all times when not in use, being tested or being maintained. If the master transfer switch is moved to the transfer position there is an alarm in the control room.

Bases 4.13:

Fire detectors are tested in accordance with the manufacturer's recommendations. All tests and inspections are performed by the plant staff. Every six months each detector is functionally tested. Combustion generated smoke is not used in these tests. Alarm circuits are functionally checked every six months. In addition, all circuitry is automatically supervised for open wiring and ground faults.

Fire pumps are tested each month to verify operability. Test starting of the screen wash/fire pump is not required since it is normally in service. Each fire pump is manually started and operated for at least 15 minutes with pump flow directed through the recirculation test line. Every 18 months the operability of the automatic actuation logic for the fire pumps and the screen wash/fire pump is verified and the performance of each pump is verified to meet system requirements. The specified flush and valve checks provide assurance that the piping system is capable of supplying fire suppression water to all safety related areas.

A system flow test is specified every three years. This test verifies the hydraulic performance of the fire suppression fire water header system. The testing will be performed using Section II, Chapter 5 of the Fire Protection Handbook, 14th Edition, as a procedural guide. This test is generally performed in conjunction with a visit from insurance company inspectors.

Hose stations and yard hydrant hose houses are inspected monthly to verify that all required equipment is in place. Gaskets in hose couplings are inspected periodically and the hose is pressure tested. Pressure testing of outdoor hose is conducted more frequently than indoor hose because of the less favorable storage conditions. Operability of hose station isolation valves is verified every three years by partially opening each valve to verify flow. All of these tests provide a high degree of assurance that each hose station and yard hydrant hose house will perform satisfactorily after periods of standby service.

Simulated automatic actuation tests are conducted each 18 months to confirm the operability of the sprinkler and Halon systems. These tests consist of verification that all valves, dampers (Halon system only), alarms, and flow paths are functional.

Plant fire barrier walls are provided with seals for pipes and cables where necessary. Where such seals are installed, they must be maintained intact to perform their function. Visual inspection of each installed seal is required every 18 months and after seal repair. A visual inspection following repair of a seal is sufficient to assure that seal integrity will be within acceptable limits.

Bases 4.13 (Continued):

Once per operating cycle the master transfer switch is moved to the transfer mode and it is verified that an alarm in the control room is received, notifying operators that control has been transferred. In addition, once per cycle, each switch is functionally tested to assure that the alternate shutdown system panel is operable and can control those systems contained to perform their design function. A frequency of more than once per operating cycle could adversely impact safety as control is taken from the normal position in the control room and the automatic initiation signals are disabled.

Bases 3.14/4.14:

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 Learned Task Force Status Report and Short Term Recommendations".

Bases 3.15/4.15:

The inservice inspection program for the Monticello plant conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection of components classified into NRC Quality Groups A, B, and C conforms to the requirements of ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code. A program of inservice testing of Quality Group A, B, and C pumps and valves is also in effect at the Monticello plant, that conforms to the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code or where alternate testing is justified in accordance with Generic Letter 89-04. If a Code required inspection is impractical for the Monticello facility, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10-year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

Bases 3.16/4.16:

A. Sample Collection & Analysis

The Radiation Environmental Monitoring Program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the plant operation. This program thereby supplements the radiological effluent monitoring by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. After a specific program has been in effect for at least three years of operation, program changes may be initiated based on this experience.

The detection capabilities required by Table 4.16.2 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirement of 40 CFR 141.

B. Land Use Census

This specification is provided to ensure that changes in the use of offsite areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

C. Interlabotatory Comparison Program

The requirement for participation in an interlabotatory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as a part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

Bases 3.17:

A. Control Room Ventilation System

The Control Room Ventilation System provides air conditioning and heating as required to maintain a suitable environment in the main control room and portions of the first and second floors of the Emergency Filtration Train (EFT) building. The main control room is normally slightly pressurized and it is possible to have 0 to 100% recirculation of conditioned air. The system is designed to maintain a nominal temperature of 78°F dry bulb and 50% nominal relative humidity in the main control room in the summer and a nominal temperature of 72°F in the winter. The Control Room Ventilation System may be isolated from unfiltered external air supply by manual action.

All toxic substances which are stored onsite or stored/shipped within a 5 mile radius of the plant have been analyzed for their affect on the control room operators. It has been concluded that the operators will have at least two minutes to don protective breathing apparatus before incapacitation limits are exceeded. For toxic substance which are transported on highways within 5 miles of the plant, it has been determined that the probability of a release from the plant due to incapacitation of the operators caused by a spill is sufficiently low that this scenario may be excluded. Protection for toxic chemicals is provided through operator training.

B. Control Room Emergency Filtration System

The Control Room Emergency Filtration System assures that the control room operators will be adequately protected against the effects of radioactive leakage which may by-pass secondary containment following a loss of coolant accident or radioactive releases from a steam line break accident. The system is designed to isolate and slightly pressurize the control room on a radiation signal in the ventilation air. Two completely redundant trains are provided.

Each train has a filter unit consisting of a prefilter, HEPA filters, and charcoal adsorbers. The HEPA filters remove particulates from the Control Room pressurizing air and prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to remove any radioiodines from the pressurizing air. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample results should indicate a radioactive methyl iodide removal efficiency of a least 98% under test conditions similar to expected accident conditions. System flows should be near their design values. The verification of these performance parameters combined with the qualification testing conducted on new filters and adsorbers provide a high level of assurance that the Emergency Filtration System will perform as predicted in reducing doses to plant personnel below those level stated in Criterion 19 of Appendix A to 10 CFR 50.

Dose calculations have been performed for the Control Room Emergency Filtration System which show that, assuming 90% standby gas treatment system adsorption and filtration efficiency and 90% control room emergency filtration system adsorption and filtration efficiency and radioiodine plateout, whole body and organ doses remain within the NRC guidelines of 5 rem and 30 rem, respectively.

Bases 4.17:

A. Control Room Ventilation System

Control room air temperature is checked each shift to ensure that the continuous duty rating for the instrumentation and equipment cooled by this system is not exceeded.

Demonstrating automatic isolation of the control room using simulated accident signals assures control room isolation under accident conditions.

B. Control Room Emergency Filtration System

Air flow through the filters and charcoal adsorbers each month assures operability of the system.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber tray is installed which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration the system adsorbs though its life. Sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for the methyl iodide removal efficiency tests. Each module withdrawn will be replaced or blocked off. In-place testing procedures will be established utilizing applicable sections of Regulatory Guide 1.52, Revision 2 and ANSI N510-1980 standards as procedural guidelines only. If test results are unacceptable, all adsorbent in the train is replaced. Any HEPA filters found defective are replaced.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 8 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

Demonstrating automatic control room pressurization using simulated accident signals assures control room pressurization with respect to adjacent areas under accident conditions.