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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

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US Nuclear Regulatory Commission
Washington , DC 20555
Attn: Document Control Desk

Tech Specification 6.8.K

Monticello Nuclear Generating Plant
Docket No. 50-263 License No. DPR-22

Technical Specification Bases Pages

Using the Monticello Technical Specification Bases Control Program, Monticello Technical Specification Bases pages have been changed. Affected pages are designated with the amendment applicable at the time and the suffix "a." The changes are summarized in Attachment A. Marked up pages applicable at the time the changes were made are provided in Attachment B. A final typed copy of the changed pages which are still applicable, for entry into the NRC authority copy, is provided in Attachment C. The current copy of our list of effective pages and record of revision is attached for your information, as Attachment D.

Please contact Paul Hartmann, at 763-271-5172 with any question or comments.

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Senior Vice President
Monticello Nuclear Generating Plant

Attachment A – Summary of Technical Specification Bases Changes (TSBC)
Attachment B – Monticello Technical Specification Bases Pages Marked Up With Changes
Attachment C – Revised Monticello Technical Specification Bases Pages
Attachment D – Monticello Technical Specification List of Effective Pages and Record of Revision

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Attachment A

Summary of Technical Specification Bases Changes (TSBC)

Following is a summary of the bases changes forwarded herein. All changes have been processed in accordance with the Monticello Technical Specification Bases Control Program described in Technical Specification 6.8.K.

TSBC-115a

Technical Specification (TS) Involved – 3.17.A
Page affected – 229y
Summary of Change: This Bases Change reflected the addition of a control room breathing air supply system.

TSBC-118a

TS Involved – 3.6/4.6
Page affected – 145
Summary of Change: This Bases Change was in response to a change for nuclear analysis to recognize a 50 degree F limitation between an idle recirculation loop and an operating recirculation loop.

TS Involved – 3.2
Page affected – 69
Summary of Change: Clarified language regarding transfer of power.

TS Involved – 3.5.1.C
Page affected – 112
Summary of Change: The discussion of a containment spray/cooling subtrain was changed to 1 RHR pump, 1 RHR Service Water Pump, and 1 RHR heat exchanger, from 2 RHR pumps, 2 RHR Service Water Pumps and 1 RHR Heat Exchanger.

Note: When TS 3.5.1.C was changed, the Bases was not updated at that time requiring a separate Bases change. This change was made but removed due to a License amendment inserting non-updated Bases language (see Bases change 124a below).

TSBC-119a

TS Involved – 3.2
Page affected – 66
Summary of Change: Revise the Basis to adopt Improved Technical Specification Bases (3.3.6.1) for the Primary Containment Isolation System Group 1 isolation signal on low steam line pressure.

TSBC-122a

TS Involved – 2.4
Pages affected – 24 and 25
Summary of Changes:
Page 24: Removed typographical error (period).
Page 25: Removed redundant sentence regarding high pressure scram and safety relief valve operability.

TSBC-123a

TS Involved – 3.7.A
Pages affected – 179 and 180
A plant modification changed the vacuum breaker position indicating lights, the Bases change reflected the modified indicating system.

TSBC-124a

TS Involved – 3.5.1.C
Page affected – 112
Summary of Change: The discussion of a containment spray/cooling subtrain was changed to 1 RHR pump, 1 RHR Service Water Pump, and 1 RHR heat exchanger, from 2 RHR pumps, 2 RHR Service Water Pumps and 1 RHR heat exchanger. When TS 3.5.1.C was changed, the bases was not updated at that time.

Attachment B

Monticello Technical Specification Bases Pages
Marked Up With Changes

This attachment consists of Monticello Technical Specification bases pages marked up with changes. The pages included are listed below:

Page

24
25
66
69
112
112
145
179
180
229y

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, with at least one safety/relief valve set to open at a pressure no greater than 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 1775 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 five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig 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. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

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 (1109 psig + 1%) or lower. However, the as-found set point can be as much as 22.3 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 3.2 (Continued):

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).~~

See Insert B

The RWCU high flow and temperature instrumentation is provided to detect a break in the RWCU piping. Tripping of this instrumentation results in actuation of the RWCU isolation valves, i.e., Group 3 valves. The trip settings have been established so that the radiological consequences of a high energy line break in this system are bounded by a break in the main steam system. The recirc sample isolation valves, which receive a Group 1 isolation signal, also receive a redundant Group 3 isolation signal.

Attachment 8

11a

Insert – Proposed Wording

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low Function is directly assumed in the analysis of the pressure regulator failure (USAR Section 7.6.3.2.4-4). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.A is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE during power operation since this is when the assumed transient can occur (USAR Section 7.6.3.2.4-4).

This Function isolates the Group 1 valves.

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

On loss of voltage, the voltage sensing relays trip immediately and energize auxiliary relays that control the bus transfer sequence.

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 14 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 systems. The isolation valves are opened during a cooldown to establish recirculation flow through the RHR suction line and return lines, thereby ensuring a uniform cooldown of this piping. 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 pump~~s~~, a RHR heat exchanger, ~~2~~ RHR pump~~s~~, 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):

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.

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 14 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 systems. The isolation valves are opened during a cooldown to establish recirculation flow through the RHR suction line and return lines, thereby ensuring a uniform cooldown of this piping. 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~~ 1 RHR service water pump~~s~~, a RHR heat exchanger, ~~2~~ 1 RHR pump~~s~~ 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.

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.

Requiring the coolant temperature in an idle loop to be within 50 deg F of the reactor coolant temperature before the pump is started assures that the positive reactivity inserted by starting the idle loop will not cause the fuel to exceed applicable limits and that the change in coolant temperature at the reactor nozzles and bottom head region are within conditions analyzed for the reactor vessel thermal and pressure transients.

Bases 3.7 (Continued):

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~~ ON
	2 Red	- ~~Off~~ ON
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~~ red 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~~ green 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.

Self contained breathing apparatus (SCBAs) and the Control Room Breathing Air Supply.

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 system is designed to maintain a nominal temperature of 78°F dry bulb in the main control room in the summer and a nominal temperature of 72°F in the winter. During normal operation, the CRV system recirculates the air in the control room envelope as needed. During a high radiation event, the Control Room Ventilation System continues to operate, and the Control Room Emergency Filtration Train system will start automatically to pressurize the control room protective envelope. The Emergency Filtration Train system can also be started manually.

All toxic substances which are stored onsite or stored/shipped within a 5 mile radius of the plant have been analyzed for their effect 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, steam line break accident or fuel handling accident. The system is designed to 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 verification of 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 levels stated in Criterion 19 of Appendix A to 10 CFR 50. The allowable penetration for the laboratory test is based on a conservative adsorber efficiency of 99% and a safety factor of ≥2.

Dose calculations have been performed for the Control Room Emergency Filtration System which show that, assuming 85% standby gas treatment system overall removal efficiency and 98% control room emergency filtration system overall removal efficiency and radioiodine plateout, whole body and organ doses remain within NRC guidelines.

Attachment C

Revised Monticello Technical Specification Bases Pages

This attachment consists of revised Monticello Technical Specification Bases pages that incorporate the changes. These pages should be entered into the NRC Authority copies of Technical Specifications. The pages included are listed below:

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, with at least one safety/relief valve set to open at a pressure no greater than 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 1775 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 five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig 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. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

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 (1109 psig + 1%) or lower. However, the as-found set point can be as much as 22.3 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.

Bases 3.2 (Continued):

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.

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (USAR Section 7.6.3.2.4-4). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.A is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing power to <25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detection low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE during power operation since this is when the assumed transient can occur (USAR Section 7.6.3.2.4-4).

This Function isolates the Group 1 valves.

The RWCU high flow and temperature instrumentation is provided to detect a break in the RWCU piping. Tripping of this instrumentation results in actuation of the RWCU isolation valves, i.e., Group 3 valves. The trip settings have been established so that the radiological consequences of a high energy line break in this system are bounded by a break in the main steam system. The recirc sample isolation valves, which receive a Group 1 isolation signal, also receive a redundant Group 3 isolation signal.

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, the voltage sensing relays trip immediately and energize auxiliary relays that control the bus transfer sequence. 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 a steady state LOCA load that maintains adequate voltage at the 480V essential MCCS. 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 3.5/4.5 (Continued):

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.

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 14 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 systems. The isolation valves are opened during a cooldown to establish recirculation flow through the RHR suction line and return lines, thereby ensuring a uniform cooldown of this piping. 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 1 RHR service water pump, a RHR heat exchanger, 1 RHR pump, 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.

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 positive reactivity inserted by starting the idle loop will not cause the fuel to exceed applicable limits and that the change in coolant temperature at the reactor 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.

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	- On
	2 Red	- On
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

Bases 3.7 (Continued):

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 red 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 green 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.

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 system is designed to maintain a nominal temperature of 78°F dry bulb in the main control room in the summer and a nominal temperature of 72°F in the winter. During normal operation, the CRV system recirculates the air in the control room envelope as needed. During a high radiation event, the Control Room Ventilation System continues to operate, and the Control Room Emergency Filtration Train system will start automatically to pressurize the control room protective envelope. The Emergency Filtration Train system can also be started manually.

All toxic substances which are stored onsite or stored/shipped within a 5 mile radius of the plant have been analyzed for their effect on the control room operators. It has been concluded that the operators will have at least two minutes to don breathing apparatus before incapacitation. Protection for toxic chemicals is provided through operator training, self-contained breathing apparatus (SBCAs) and the Control Room Breathing Air Supply.

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, steam line break accident or fuel handling accident. The system is designed to 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 verification of 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 levels stated in Criterion 19 of Appendix A to 10 CFR 50. The allowable penetration for the laboratory test is based on a conservative adsorber efficiency of 99% and a safety factor of ≥ 2.

Dose calculations have been performed for the Control Room Emergency Filtration System which show that, assuming 85% standby gas treatment system overall removal efficiency and 98% control room emergency filtration system overall removal efficiency and radioiodine plateout, whole body and organ doses remain within NRC guidelines.

Attachment D

Monticello Technical Specification
List of Effective Pages and Record of Revision

This attachment consists of the current Monticello Technical Specification List of Effective Pages and Record of Revision. The pages included are listed below:

Page

A through I

MONTICELLO NUCLEAR GENERATING PLANT
APPENDIX A TECHNICAL SPECIFICATIONS RECORD OF REVISIONS

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B	124		29	83	66	119a	110	100a	
C	115		30	103	67	117	111	122	
D	115		31	104	68	102	112	124a	|
E	115		32	103	69	118a	113	122	
F	115		33	103	69a	100a	114	122	
G	115		34	83	70	117	121	0	
H	119		35	100a	71	100a	122	106	
I	125	|	36	100a	71a	105	123	117	
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ii	104		38	102	76	0	125	104	
iii	120		39	102	77	86	126	104	
iv	124		40	100a	78	0	126a	87	
v	120		42	103	79	0	127	122	
vi	121		45	0	80	29	128	42	
vii	122		46	70	81	3	129	122	
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2	70		47	40	82a	63	131	122	
3	21		48	89	83	24	132	39	
4	102		49	102	83a	24	132a	122	
5	120		50	117	84	100a	133	106	
5a	120		50a	117	85	100a	134	106	
6	125	|	51	117	86	100a	135	106	
7	29		51a	117	87	100a	136	106	
8	29		52	103	88	100a	137	0	
10	100a		53	103	89	104	138	100a	
11	100a		54	103	90	100a	145	118a	
12	100a		55	103	91	123	146	106	
13	100a		56	102	92	100a	147	107	
14	102		57	70	93	122	148	117	
15	102		58	84	94	106	149	100a	
16	102		58a	29	95	77	150	102	
18	100a		59	103	96	77	151	114	
19	102		59a	103	97	57	153	100a	
20	100a		60	63	98	56	154	100a	
21	43		60a	31	99	104	155	122	
22	104		60b	62	100	100a	156	93	
23	102		60c	30	101	122	157	117	
24	122a		60d	105	102	122	158	107	
25	122a		60e	89	103	122	159	95	
25a	115		61	104	104	122	160	95	
25b	122		62	117	105	122	163	0	
26	5		63	117	106	79	164	104	
27	81		63a	117	107	97	165	64	
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177	117	229d	63
178	100a	229e	122
179	123a	229u	104
180	123a	229v	112
181	100a	229v v	112
182	112	229w	112
183	117	229ww	112
184	100a	229x	112
185	100a	229y	115a
188	104	229z	112
189	100a	230	54
190	104	231	34
191	0	232	119
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197	121	236	115
198	121	243	115
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200	104	248	59
201	77	249	120
202	80	250	120
203	41	251	124
204	100a	252	120
205	100a	253	120
206	0	254	120
207	123	255	120
208	63	256	122
209	123	257	122
209a	100a	258	122
210	100a	259	120
211	109	260	120
212	109	261	120
213	99	262	120
216	100a		

MONTICELLO NUCLEAR GENERATING PLANT
RECORD OF TECHNICAL SPECIFICATION CHANGES AND LICENSE AMENDMENTS

NSP Page Revision (REV) No.	License DPR-22 Amend No. & Date	AEC Tech Spec Change Issuance No. and date	Major Subject
Original	-	-	Appendix A Technical Specifications incorporated in DPR-22 on 9/8/70
-	1 1/19/71	Note 1	Removed 5 MWt restriction
-	Note 2	2 1/14/72	MOGS Technical Specification changes issued by AEC but never distributed or put into effect, superseded by TS Change 12 11/15/73
1	Note 2	3 10/31/72	RHR service water pump capability change
-	Note 2	4 12/8/72	Temporary surveillance test waiver
-	2 2/20/73	Note 1	Increase in U-235 allowed in fission chambers
2	Note 2	5 3/2/73	Miscellaneous Technical Specification changes,
3	Note 2	1 4/28/71& 6 4/3/73	Respiratory Protection, & Administrative Control Changes
4	Note 2	7 5/4/73	Respiratory Protection Changes
5	Note 2	8 7/2/73	Relief Valve and CRD Scram Time Changes
6	Note 2	9 8/24/73	Fuel Densification Limits
7	Note 2	10 10/2/73	Safety Valve Setpoint Change
8	Note 2	11 11/27/73& 12 11/15/73	Offgas Holdup System, RWM, and Miscellaneous Changes
9	Note 2	13 3/30/74	8x8 Fuel Load Authorization
10	3	14 5/14/74	8x8 Full Power authorization
-	4 6/17/74	Note 1	Changed byproduct material allowance
-	6 8/20/74	Note 1	Changed byproduct material allowance
11	Note 3	Note 3 10/24/74	Inverted Tube (CRD) Limits
12	5	15 1/15/75	REMP Changes
13	7	16 2/3/75	Reactor Vessel Surveillance Program Changes
14	8	17 2/26/75	Vacuum Breaker Test Changes
15	9	18 4/10/75	Corrects Errors & Provides Clarification
-	10 7/8/75	Note 1	Increased allowed quantity of U-235
16	12	20 9/15/75	Snubber Requirements
17	11	19 9/17/75	Removed byproduct material allowance
18	13	21 10/6/75	Suppression Pool Temperature Limits
19	14	22 10/30/75	Appendix K and GETAB Limits
20	15 1/22/76 NOTE 4		Reporting Requirements
21	16 2/3/76		CRD Collet Failure Surveillance
22	17 3/16/76		NSP Organization Changes
23	NOTE 3 4/13/76		Adoption of GETAB
24	18 4/14/76		Containment Isolation Valve Testing
25	21 5/20/76		Interim Appendix B, Section 2.4 Tech. Specs.

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26	19	5/27/76	Low Steamline Pressure Setpoint and MCPR Changes
27	20	6/18/76	APLHGR, LHGR, MCPR Limits
28	22	7/13/76	Correction of Errors and Environmental Reporting
29	23	9/27/76	Standby Gas Treatment System Surveillance
30	24	10/15/76	CRD Test Frequency
31	25	10/27/76	Snubber Testing Changes
32	26	4/1/77	APRS Test Method
33	27	5/24/77	MAPLHGR Clamp at Reduced Flow
34	28	6/10/77	Radiation Protection Supervisor Qualification
35	29	9/16/77	REMP Changes
36	30	9/28/77	More Restrictive MCPR
37	31	10/14/77	Inservice Inspection Changes
38	32	12/9/77	Reporting Requirements
39	33	1/25/78	Fire Protection Requirements
NOTE 1	34	4/14/78	Increase in spent fuel storage capacity
40	35	9/15/78	RPT Requirements
41	36	10/30/78	Suppression Pool Surveillance
42	37	11/6/78	8x8R Authorization, MCPR Limits & SRV Setpoints
43	NOTE 3	11/24/78	Corrected Downcomer Submergence
44	38	3/15/79	Incorporation of Physical Security Plan into License
45	39	5/15/79	Revised LPCI Flow Capability
46	40	6/5/79	Respiratory Protection Program Changes
47	41	8/29/79	Fire Protection Safety Evaluation Report
48	42	12/28/79	MAPLHGR vs. Exposure Table
49	43	2/12/80	MCPR & MAPLHGR Changes for Cycle 8 and Extended Core Burnup
50	44	2/29/80	ILRT Requirements
NOTE 1	-	8/29/80	Order for Modification of License-Environmental Qualification
NOTE 1	-	9/19/80	Revised Order for Modification of License-Environmental Qualification
51	-	10/24/80	Order for Modification of License-Environmental Qualification Records
52	-	1/9/81	Issuance of Facility Operating License (FTOL)
NOTE 1	-	1/9/81	Order for Modification of License Concerning BWR Scram Discharge Systems (License conditions removed per Amendment No. 11 dated 10/8/82)

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NSP Revision (REV) No.	License DPR-22 Amend No.	& Date	Major Subject
NOTE 1	-	1/13/81	Order for Modification Mark I Containment
-	1	2/12/81	Revision of License Conditions Relating to Fire Protection Modifications
53	2	3/2/81	TMI Lessons Learned & Safety - Related Hydraulic Snubber Additions
54	3	3/27/81	Low voltage protection, organization and miscellaneous
NOTE 1	4	3/27/81	Incorporation of Safeguards Contingency Plan and Security Force Qualification and Training Plan into License
55	5	5/4/81	Cycle 9 - ODYN Changes, New MAPLHGR's, RPS Response time change
56	6	6/3/81	Inservice Inspection Program
57	7	6/30/81	Fire Protection Technical Specification Changes
58	8	11/5/81	Mark I Containment Modifications
59	9	12/28/81	Inservice Surveillance Requirements for Snubbers
NOTE 1	-	1/19/82	Revised Order for Modification Mark I Containment
60	10	5/20/82	Scram Discharge Volume
61	11	10/8/82	New Scram Discharge Volumes
62	12	11/30/82	RPS Power Monitor
63	13	12/6/82	Cycle 10
64	14	12/10/82	Recirc Piping and Coolant Leak Detection
65	15	12/17/82	Appendix I Technical Specifications (removed App. B)
66	16	4/18/83	Organizational Changes
67	17	4/17/83	Miscellaneous Changes
68	18	11/28/83	Steam Line Temperature Switch Setpoint
69	19	12/30/83	Radiation Protection Program
70	20	1/16/84	SRM Count Rate
71	21	1/23/84	Definition of Operability
72	22	2/2/84	Miscellaneous Technical Specification Changes
73	23	4/3/84	RPS Electrical Protection Assembly Time Delay
74	24	5/1/84	Scram Discharge Volume Vent and Drain Valves
75	25	8/15/84	Miscellaneous Technical Specification Changes
76	26	9/24/84	Cycle 11
77	27	10/31/84	RHR Intertie Line Addition
78	28	11/2/84	Hybrid I Control Rod Assembly
79	29	11/16/84	ARTS
80	30	11/16/84	Low Low Set Logic
81	31	11/27/84	Degraded Voltage Protection Logic

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82	32	5/28/85	Surveillance Requirements
83	33	10/7/85	Screen Wash/Fire Pump (Partial)
84	34	10/8/85	Fuel Enrichment Limits
85	35	12/3/85	Combustible Gas Control System
86	36	12/23/85	Vacuum Breaker Cycling
87	37	1/22/86	NUREG-0737 Technical Specifications
88	38	2/12/86	Environmental Technical Specifications
89	39	3/13/86	Administrative Changes
90	40	3/18/86	Clarification of Radiation Monitor Requirements
91	41	3/24/86	250 Volt Battery
92	42	3/27/86	Jet Pump Surveillance
93	43	4/8/86	Simmer Margin Improvement
94	44	5/27/86	Cycle 12 Operation
95	45	7/1/86	Miscellaneous Changes
96	46	7/1/86	LER Reporting and Miscellaneous Changes
97	47	10/22/86	Single Loop Operation
98	48	12/1/86	Offgas System Trip
99	49	8/26/87	Rod Block Monitor
100	50	8/26/87	APRM and IRM Scram Requirements
101	51	10/16/87	2R Transformer
102	52	11/18/87	Surveillance Intervals - ILRT Schedule
103	53	11/19/87	Extension of Operating License
104	54	11/25/87	Cycle 13 and Misc Changes
105	55	11/25/87	Appendix J Testing
106	56	12/11/87	ATWS - Enriched Boron
107	57	9/23/88	Increased Boron Enrichment
108	58	12/13/88	Physical Security Plan
109	59	2/16/89	Miscellaneous Administrative Changes
110	60	2/28/89	Miscellaneous Administrative Changes
111	61	3/29/89	Fire Protection and Detection System
112	62	3/31/89	ADS Logic and S/RV Discharge Pipe Pressure
113	63	4/18/89	Miscellaneous Technical Specification Improvements
114	64	5/10/89	Containment Vent and Purge Valves
115	65	5/30/89	NUREG-0737 - Generic Letter 83-36
116	66	5/30/89	Reactor Vessel Level Instrumentation

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NSP Revision (REV) No.	License DPR-22 Amend No. & Date		Major Subject
117	67	6/19/89	Extension of MAPLHGR. Exposure for One Fuel Type
118	68	7/14/89	SRO Requirements & Organization Chart Removal
119	69	9/12/89	Operations Committee Quorum Requirements
120	70	9/28/89	Relocation of Cycle-Specific Thermal-Hydraulic Limits
121	71	10/19/89	Deletion of Primary Containment Isolation Valve Table
122	72	11/2/89	RG 1.99, Rev 2, ISI & ILRT
123	73	5/1/90	Combined STA/LSO Position
124	74	6/5/90	Removal of WRGM Automatic ESF Actuation
125	75	10/12/90	Diesel Fuel Oil Storage
126	76	12/20/90	Miscellaneous Administrative Changes
127	77	2/15/91	Redundant and IST Testing
128	78	3/28/91	Alarming Dosimetry
125	79	4/9/91	SAFER/GESTR
130	80	8/12/91	Torus Vacuum Breaker Test Switch/EDG Fuel Oil Tank Level
131	81	4/16/92	Surveillance Test Interval Extension - Part I
132	82	7/15/92	Alternate Snubber Visual Inspection Intervals
133	83	8/18/92	Revisions to Reactor Protection System Tech Specs
134	84	1/27/93	MELLIA and Increase Core Flow
135	85	6/29/93	Revision to Diesel Fire Pump Fuel Oil Sampling Requirements
136	86	7/12/93	Revisions to Control Rod Drive Testing Requirements
137	87	4/15/94	Revised Coolant Leakage Monitoring Frequency
138	88	6/30/94	Average Planar Linear Heat Generation Rate (APLHGR) Specification & Minimum Critical Power Ratio Bases Revisions
139	89	8/25/94	Removal of Chlorine Detection Requirements and Changes to Control Room Ventilation System Requirements
140	90	9/7/94	Revisions to Radiological Effluent Specifications
141	91	9/9/94	Secondary Containment System and Standby Gas Treatment System Water Level Setpoint Change
142	92	9/15/94	Change in Safety Relief Valves Testing Requirements
143	93	7/12/95	Revised Core Spray Pump Flow
144	94	10/2/95	Standby Gas Treatment and Secondary Containment Systems
145	95	4/3/96	MSIV Combined Leakrate, and Appendix J, Option B
146	96	4/9/96	Purge and Vent Valve Seal Replacement Interval
147	97	9/17/96	Implementation of BRWOG Option I-D core Stability Solution and re-issue of pages 11, 12, 82 and 231 to reflect pages issued by NRC amendments.

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NSP Revision (REV) No.	License DPR-22 Amend No.	& Date	Major Subject
148	98	7/25/97	Bases changes on containment overpressure and number of RHR pumps required to be operable. Reissue pages 207, 209, 219, 229k, 229p, 230, 245 to reflect pages issued by NRC amendments.
149	99	10/29/97	SLMCPR for Cycle 18 and reissue pages vi, 155, 202, 207, 219, 229u
NOTE 5		11/25/97	Reissue pages a, b, g, iii, vi, 14, 25a, 155, 198y, 198z, 202, 207, 209, 219, 229k, 229p, 229r, 229u, 230, 245
150	100	4/20/98	SLMCPR for Cycle 19
NOTE 6	100a	4/30/98	Reissue all pages.
	101	8/28/98	Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability
	102	9/16/98	Monticello Power Rerate
	103	12/23/98	Surveillance Test Interval/Allowed Outage Time Extension Program - Part 2
	104	12/24/98	Revision of Statement on Shift Length & other Misc Changes
	105	03/19/99	CST Low Level HPCI/RCIC Suction Transfer
	106	10/12/99	Revised RPV-PT Curves & remove SBLC RV setpoint
	107	11/24/99	Reactor Pressure Vessel Hydrostatic and Leakage Testing
	108	12/8/99	Testing Requirements for Control Room EFT Filters
	109	2/16/00	Safety Limit Minimum Critical Power Ratio for Cycle 20
	110	8/7/00	Transfer of Operating Authority from NSP to NMC
	111	8/18/00	Transfer of Operating License from NSP to a New Utility Operating Company
	112	8/18/00	Emergency Filtration Train Testing Exceptions and Technical Specification Revisions
	113	10/2/00	Alternate Shutdown System Operability Requirements
	114	11/30/00	Safety/Relief Valve Bellows Leak Detection System Test Frequency
	115	12/21/00	Administrative Controls and Other Miscellaneous Changes
	115a	02/13/01	Bases Change to Reflect Modification 98Q145 Installed Control Room Toxic Gas Air Supply
	116	03/01/01	Relocation of Inservice Inspection Requirements to a Licensee Program
	117	03/07/01	Reactor Water Cleanup (RWCU) System Automatic Isolation and Miscellaneous Instrumentation System Changes
	118	03/09/01	Revision of Standby Liquid Control System Surveillance Requirements
	118a	05/10/01	Bases Change - 50°F Loop Temperature, Bus Transfer & Rerate Correction

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NSP Revision (REV) No.	License DPR-22 Amend No. & Date		Major Subject
	119	04/05/01	Fire Protection Technical Specification Changes
	119a	06/28/01	Bases Change - Added information on cooldown rate
	120	07/24/01	Relocation of Radiological Effluent Technical Specifications to a Licensee-Controlled Program
	121	07/25/01	Clarify air ejector offgas activity sample point and operability requirements
	122	08/01/01	Relocation of Inservice Testing Requirements to a Licensee-Controlled Program
	122a	10/22/01	Bases Change - Remove scram setpoints sentence and correct typo
	123	10/26/01	Control Rod Drive and Core Monitoring Technical Specification Changes
	123a	10/25/01	Bases Change - Change to reflect new operation of drywell to suppression chamber vacuum breaker valve position indicating lights
	124	10/30/01	Relocation of Technical Specification Administrative Controls Related to Quality Assurance Plan
	124a	12/05/01	Bases Change - Change to reflect revised Technical Specification definition of a containment spray/cooling subsystem
	125	12/06/01	Safety Limit Minimum Critical Power Ratio for Cycle 21

1. License Amendment or Order for Modification of License not affecting Technical Specifications.
2. Technical Specification change issued prior to 10 CFR revisions which require issuance of Technical Specification changes as License Amendments.
3. Modification to Bases. No Technical Specification change or License Amendment issued.
4. Technical Specification change numbers no longer assigned beginning with Amendment 15.
5. Pages reissued 11/25/97 to conform with NRC version. Revision number of effected pages not changed.
6. All pages reissued using INTERLEAF in different font. Using NRC Amendment Nos. and issue date. For Bases and Table of Contents, spelling errors corrected and editorial corrections made and all Amendment Nos. changed to 100a. For remaining Tech Spec pages, no other changes made and current Amendment Nos. used.