EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS

The following sections include a brief description and a summary of the 10CFR50.59 evaluations for those changes, tests and experiments that were carried out without prior NRC approval, pursuant to the requirements of 10 CFR Part 50, Section 50.59(d)(2).

1. <u>Domestic Water Supply to Plant Systems (SRI 00-003)</u>

<u>Description</u>:

The Safety Review Item (SRI) evaluated proposed revisions to Section 10.3.5 of the Updated Safety Analysis Report (USAR) that clarified and provided additional description of the Domestic Water System. Previously, the USAR stated that the Domestic Water System is completely independent of the plant process systems, which was incorrect. It is connected to both the Seal Water System and the Plant Makeup Water Treatment System. This SRI was required because it provided the technical and regulatory justification for the proposed USAR changes.

<u>10CFR50.59 Evaluation Summary</u>:

The description of the Domestic Water System was clarified in USAR Sections 10.3.5.1, 10.3.5.2 and 10.3.5.3. The clarifications state that this system provides raw water to the Plant Makeup Water Treatment System and normal supply for the Seal Water System. These clarifications also provide additional description for the Seal Water supply and a USAR reference for the Plant Makeup Water Treatment System. The revision deleted the incorrect statement that the Domestic Water System is completely independent of the plant process systems. These changes had no effect on the design, function or operation of this system.

2. <u>Automatic Depressurization Systems (ADS) Circuit Description Discrepancies (SRI 98-009)</u>

<u>Description</u>:

This 10CFR50.59 evaluation addressed discrepancies noted in actuation logic and other circuit descriptions between ADS system prints and the USAR description. An NRC resident inspection identified and addressed an incorrect description of the Automatic Depressurization System (ADS) logic in the USAR. Two USAR corrections resulted in the ADS logic circuit being referenced as "*two-out-of-two-once*" rather than "*one-out-of-two twice*." The change was consistent with ADS system prints, Technical Specification Table 3.2.2 and Technical Specification Bases. Two prudent clarifications better described an AC-interlock permissive and an automated power supply switchover scheme. Also, a statement about certain switches being permissive while de-energized was eliminated. This SRI was required to address the non-conforming conditions by performance of a 10CFR50.59 evaluation.

10CFR50.59 Evaluation Summary:

These USAR changes were necessary to make the ADS description in Section 6.2.5.2 reflect the actual system configuration. The statement referencing "permissive when de-energized" switches was removed. The ADS actuation logic descriptions were changed to agree with station prints and Technical Specifications Bases and Table 3.2.2 by changing the "*one-out-of-two-twice*" logic statements to "*two-out-of-two once*." Two other clarifications better described the power bus transfer scheme and the AC interlock logic associated with the operation of the low pressure pumps. These changes did not represent an unreviewed safety question and provided a more accurate and clearly written USAR.

3. Control Room Ventilation Continuous Use of EFT for Outside Air (SRI 00-016)

Description:

This 10CFR50.59 evaluation addressed the practice of the Control Room Ventilation (CRV) continually drawing outside air through the Emergency Filtration Trains (EFTs) rather than an infrequent use as was described in the USAR. Also, recirculation through the air conditioning units is not the exclusive, normal mode of operation, since fresh outside air is also constantly introduced to the CRV system through the EFTs in conjunction with recirculation.

10CFR50.59 Evaluation Summary:

USAR Section 6.7 changes to incorporate constar[illegible] of the EI[illegible] for fresh air supply to the control room were acceptable with respect to safety and reliability. The change allowed the CRV system to operate in recirculation mode exclusively or in conjunction with a continuous supply of fresh air. The "normal" mode of operation is described as using both recirculation and supplemental outside air through the EFTs.

4. All Feedwater Pump Trips Not Listed in USAR 7.7.4.2 (SRI 00-004)

Description:

The USAR Review Project identified that not all of the Reactor Feedwater Pump Trips were listed in USAR Section 7.7.4.2. This SRI evaluated the addition of the low suction flow, low lube oil pressure and motor fault pump trips to this USAR section. This SRI was required because it provided the technical and regulatory justification for the proposed USAR change.

10CFR50.59 Evaluation Summary:

The USAR Review Project identified that not all of the Reactor Feedwater Pump Trips were listed in USAR Section 7.7.4.2. SRI 00-004 provided justification for also listing the low suction flow and low lube oil pressure trips, which were part of the pump motors original design, and also for listing the motor fault (overload) trip, which was added by an approved modification. This USAR revision provides a complete listing of the Reactor Feedwater Pump trips.

5. <u>Heavy Loads Definition / Limit (SRI 00-011)</u>

<u>Description</u>:

USAR Section 12.2.5.1 states in part,

A heavy load is defined, by NUREG-0612, as "any load, carried in a given area ..., that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool ...". At Monticello, a heavy load has been conservatively calculated to be any load heavier than 1500 lbs.

Based on the NUREG-0612 definition, the heavy load limit at Monticello was established at 1500 pounds. The combined weight of a single spent fuel assembly and its associated handling tool at Monticello was about 1570 pounds, which included the weight of the handling equipment (e.g., fuel bridge grapple and arm) of approximately 870 pounds and the weight of the fuel bundle of approximately 700 pounds. Based on this value, Monticello chose a more conservative limit of 1500 pounds to use as the heavy load limit. Modification 83MO91, *Refueling Bridge Mast Replacement & Bridge Upgrade,* (Ref. 1) replaced the mast and grapple assembly with a lighter stainless steel assembly. The new assembly weighs approximately 350 pounds. Therefore, the combined weight of a single spent fuel assembly and its associated handling tool is currently about 1050 pounds (i.e., 700 lbs + 350 lbs = 1050 lb).

<u>10CFR50.59 Evaluation Summary</u>:

The purpose of this SRI was to provide a description and evaluation of a USAR change that was identified via the corrective action program. The heavy load limit over the spent fuel pool and the reactor core was conservatively established at 1100 lb and the heavy load limit for the balance of plant remained at 1500 lb. The heavy load limits met the intent of NUREG-0612. They had no impact upon Technical Specifications. They did not involve changes to operation, maintenance or testing. There were no new physical or design changes. The changes met applicable codes and regulations. No new types of hazards, failure modes or interactions were identified. Existing license basis accident analyses and radiation dose calculations remain valid.

6. <u>Torus Cooling During RCIC Operation (SRI 00-005)</u>

<u>Description</u>:

This 10CFR50.59 evaluation addressed the time allowed before torus cooling becomes necessary once the Reactor Core Isolation Cooling (RCIC) turbine exhaust steam begins discharging to the suppression pool. A portion of a statement in USAR Section 10.2.5.2, regarding a four-hour time duration available before torus cooling is necessary with RCIC steam turbine operation, was identified as inaccurate. The text was non-descriptive with respect to the basis for the stated duration and did not clearly indicate whether cooling was to be initiated or the desired effects of cooling occurred at the four-hour mark. The statement was qualitative with dependence on the available heat sink and the corresponding heat production under RCIC operation, but the heat capacity of the suppression pool was variable with its initial temperature, therefore, it was meaningless to declare a specific time available before initiating pool cooling. The existing suppression pool Technical Specifications are temperature based. Temperature is the first order parameter as it governs the effectiveness and capacity of the pool to quench steam. The suppression pool and Residual Heat Removal (RHR) systems are necessary for the safe shutdown of the plant and are safety related. An SRI was necessary to correct this statement.

10CFR50.59 Evaluation Summary:

The USAR Section 10.2.5.2 words, "... after a specified time interval (4 hours)" were removed to eliminate any time dependence and/or allowance to the suppression pool cooling function. This also increased the clarity of the statement by removing the erroneous reference to a time interval that was provided with no supporting basis.

The cooling requirement, as described by Technical Specifications, is governed by temperature, which is the pool parameter most significant (assuming proper volume) with regard to quenching steam. The time to initiate cooling beyond the Technical Specification limit of 90°F is an unfounded limitation, since cooling is provided, per plant procedures, prior to reaching the temperature limit. The referenced emergency procedure is consistent with the NRC approved Emergency Procedure Guidelines.

7. CRD/CRH System Discrepancies in USAR (SRI 00-001)

Description:

The issues addressed by this 10CFR50.59 evaluation dealt primarily with Control Rod Drive/Control Rod Housing (CRD/CRH) system operating parameters and physical configurations that differed from the USAR system description. Many issues from separate USAR sections are directly related to each other since they may be attributed to a particular variation in water pressure or flow, or some other common parameter. The system is operable despite the noted discrepancies and much of the noted variation is a result of incorporating operating experience in designed flow or pressure adjustments. It is, and always has been, a Monticello Nuclear Generating Plant (MNGP) imperative to operate the system in a manner to optimize reliability and performance. All system performance and Technical Specification requirements are met by the CRD/CRH system with its present configuration and operational parameters.

10CFR50.59 Evaluation Summary:

Changes made to CRD/CRH operational parameters were the direct result of adjustments to optimize operation, made possible by operating experience. Other description changes are minor in nature and do not impact the operability of the system.

The changes corrected discrepancies between USAR Section 3.5 text and actual plant parameters and configurations. The CRD/CRH system flows, pressure and associated forces were updated to reflect current operational values. Other descriptions were adjusted to more accurately describe the existing system.

8. Design Basis for RCIC Pump Minimum Flow Bypass Valve Opening Parameter (SRI 00-010)

Description:

A site condition report evaluated a discrepancy with the stroke time for the Reactor Core Isolation Cooling (RCIC) Pump Minimum Flow Bypass Valve as stated in USAR Table 10.2-3. That assessment concluded that the stroke time had been incorrectly listed in the USAR as 15 seconds. This SRI evaluated revising this stroke time to the General Electric design specification value of 5 seconds. This SRI provided the technical and regulatory justification for the proposed USAR change.

10CFR50.59 Evaluation Summary:

Although the allowable stroke time for the RCIC Pump Minimum Flow Bypass Valve listed in USAR Table 10.2-3 was 15 seconds, assessment determined that it should have been listed as 5 seconds. This SRI provided justification for revising the stroke time to 5 seconds. This was a conservative change that did not affect the capability of RCIC to deliver design flow within 30 seconds.

9. Establish Design Temperature Gradient for Spent Fuel Storage Pool Wall (SRI 00-017)

Description:

The purpose of this SRI was to provide a description and evaluation of a USAR change that was identified via the corrective action program. USAR Section 12.2.2.1.1.f, *Thermal Loads*, stated that the design loads used to evaluate the spent fuel pool structure included a temperature gradient of 50°F through slab and walls under normal (operating) conditions. Per USAR Section 10.2.2.1, the spent fuel pool cooling operation at temperatures up to 140°F is acceptable. With the minimum bulk air temperature of 60°F for areas adjacent to the spent fuel pool structure, the temperature gradient through the slab and walls of the spent fuel pool could exceed 50°F.

10CFR50.59 Evaluation Summary:

As a result of this SRI, the spent fuel pool design temperature gradient through slab walls under operating conditions was changed from 50°F to 72°F. This change had no impact on the Technical Specifications. Operations, maintenance and testing were not affected. There were no new physical or design changes. The change met applicable codes and regulations. No new types of hazards, failure modes or interactions were identified. Existing license basis accident analyses and radiation dose calculations remain valid.

10. Incorrect Bypass Delay Time for the RMCS Shutdown Scram (SRI 00-008)

Description:

This 10CFR50.59 evaluation addressed the scram bypass timer being set for two seconds rather than ten seconds as indicated in USAR Section 7.6.1.2.8. The ten-second value dates back to the original FSAR text. The physical setting of the bypass timer in the plant is, and always has been, two seconds.

10CFR50.59 Evaluation Summary:

This SRI was written to justify the change to the Reactor Manual Control System (RMCS) bypass timer description in the USAR from ten to two seconds. There was no physical change performed since the timer is currently set at two seconds. No unreviewed safety question was created by this change.

There was no functional consequence resulting from introduction of the shutdown scram bypass earlier than ten seconds (as was indicated in the USAR). The two-second duration ensures adequate time for the trip relays to de-energize the Reactor Protection System (RPS) seal-in circuit to de-energize the CRD scram pilot valves. The automatic bypass itself is entirely passive with respect to any protective function actuation and its specific delay duration is not relied upon in any accident analysis. The control rods will have been scrammed at such time the RPS is allowed to reset. Since the bypass is a prerequisite for the reset, it is prudent that the bypass timer is set to a shorter duration than the reset timer.

11. Elimination of Design Condition Parameter from USAR Table 5.2-1, "Principal Design Parameters of Primary Containment" (SRI 00-015)

Description:

This SRI evaluated an issue found during the USAR Review Project. The "normal internal pressure" for Primary Containment was listed in USAR Table 5.2-1 as 1.75 psig. No basis for this value was identified. This SRI evaluated revision of the USAR table to show the normal internal pressure range for Primary Containment. The SRI provided the regulatory and technical justification for the proposed USAR change.

10CFR50.59 Evaluation Summary:

It was determined that the "normal internal pressure" for the Primary Containment System, listed in USAR Table 5.2-1 as 1.75 psig, was incorrect and did not have a design basis. This SRI provided justification for deleting this pressure and replacing it with the correct operating pressure range. Operating data showed that the normal internal pressure was less than 1.75 psig and that the Primary Containment high-pressure signal alarm setpoint was also less than this value. This change did not affect the design or accident mitigation capability of the Primary Containment System.

12. Investigate and Resolve USAR Statements Regarding HPCI (SRI 00-002)

Description:

This SRI evaluated an issue found during the USAR Review Project. The 10CFR50.59 evaluation assessed several discrepancies between the USAR and other design documents concerning the High Pressure Coolant Injection (HPCI) turbine design parameters and the HPCI pump discharge isolation valves' position following a turbine trip. The 10CFR50.59 evaluation provided the regulatory and technical justification for proposed USAR changes. These proposed changes affect the HPCI System, specifically the HPCI turbine and both the HPCI pump discharge inboard and outboard isolation valves.

10CFR50.59 Evaluation Summary:

Some of the HPCI turbine design parameters listed in USAR Table 6.2-3 were revised to meet the original values as specified by the HPCI turbine vendor, Terry Turbine. These changes were either insignificant or minor changes, correction of administrative errors, or corrections for consistency between different USAR sections. A USAR statement that the pump discharge valves were prevented from opening automatically whenever a turbine trip condition existed was incorrect and was deleted. There are no design or operational requirements for this. These valves are not required to close to support any safety-related function. The proposed USAR changes do not constitute an unreviewed safety question. Revision of the HPCI turbine design parameters and allowing the HPCI pump discharge valves to open automatically when a HPCI turbine trip exists were acceptable from design, operational and radiological standpoints.

13. Install Blank Flange Downstream of XR-10-4 to Prevent Leakage per Jumper 99-29 (SRI 00-025)

Description:

The Seal Vent to Open Radwaste (ORW) for 12 Recirculation Pump (XR-10-4) leaked by. A temporary modification installed a blank flange on the non-safety related, Open Radwaste side of XR-10-4. An SRI was required because the temporary modification affected a USAR figure. The temporary modification has since been removed.

10CFR50.59 Evaluation Summary:

The Seal Vent to Open Radwaste (ORW) for 12 Recirculation Pump (XR-10-4) leaked by. A blank flange was temporarily installed on the non-safety related Open Radwaste side of XR-10-4. The 50.59 evaluation was required because the temporary modification affected a USAR figure. Since the installation was a temporary modification, the affected USAR drawing was not revised. Addition of the blank flange did not constitute an unreviewed safety question.

14. Reactor Recirculation MG Set and Pump Testing (SRI 00-009)

Description:

During preparation for the reactor pressure vessel hydro a problem occurred with the 11 Reactor Recirculation pump motor generator (11 Recirc MG Set) which caused the unit to trip. To provide for post maintenance testing of the 11 Recirc MG Set, operation of the 12 Recirc MG Set, and recirculation pumps, it was necessary to operate the MG sets and pumps at other than minimum speed. This testing occurred during a refueling outage with the reactor shutdown and the reactor water temperature less than 212°F. A review was conducted to identify the potential conflicts between testing and the USAR or Technical Specifications. In order to run reactor coolant recirculation pumps at higher than minimum speed under existing plant conditions, it was necessary to temporarily bypass the <20% feedwater flow interlock. Bypassing the interlock created a conflict with the Technical Specifications on net positive suction head (NPSH) required for the recirc pump components. This SRI explained how the test was safely accomplished without violating [illegible] pump or reactor recirc pump NPSH limits during circumstances where it was de[illegible] run the recirc pumps above minimum speed.

10CFR50.59 Evaluation Summary:

The proposed activities evaluated in this 10CFR50.59 evaluation, (1) temporarily bypassing the <20% feedwater flow interlock; and (2) operating the Recirc MG sets and pumps at greater than minimum flow with less than 20% feedwater flow available, were acceptable activities and did not constitute an unreviewed safety question. These activities were performed under conditions established in this SRI and implemented under a work order, which controlled the testing of the Recirc MG sets.

15. 10CFR50.59 Evaluation for Differences Between the EOPs and the Design Basis – Defeating the HPCI High Torus Water Level Suction Transfer to Allow Continued HPCI Operation Using the CSTs as a Suction Source (SRI 99-018)

Description:

This 50.59 evaluation addressed revision of Monticello's Emergency Operating Procedures (EOPs) to defeat the HPCI high torus water level suction transfer if torus water temperature exceeds 160°F. The analyses described in the USAR assume transfer of the HPCI suction source to the suppression pool from the Condensate Storage Tank (CST) on high torus water level. This issue was evaluated as a part of a larger effort to evaluate differences between the EOPs and plant design basis.

10CFR50.59 Evaluation Summary:

Defeating the HPCI high torus water level suction transfer if torus water temperature exceeds 160°F did not invalidate the design bases or licensing basis accident analysis assumptions, did not result in consequences more severe than the consequences of taking the USAR actions, did not decrease the effectiveness of the EOPs, and did not constitute an unreviewed safety question. Revising the EOPs to defeat the HPCI high torus water level suction transfer if torus water temperature exceeds 160°F did not constitute an unreviewed safety question and did not adversely affect the ability of the EOPs to mitigate the consequences of any mechanistically credible event.

16. EOP LPCI 5-Minute Seal-in Timer Bypass Switch (Mod 00Q250)

Description:

This modification installed permanent switches to remove the need for booted relay contacts when using the following Emergency Operating Procedure (EOP) support procedures: C.5-3205 (TERMINATE AND PREVENT), C.5-3201 (DEFEAT RCIC PRESSURE AND TEMPERATURE ISOLATIONS), and C.5-3503 (DEFEAT DRYWELL COOLER TRIPS). Evaluation of bypassing the LPCI five-minute bypass timer was accomplished under SRI 01-010. Evaluation of bypassing the RCIC pressure and temperature isolations was accomplished under SRI 01-011. The evaluation of bypassing the ECCS trip of drywell cooling was accomplished under SRI 01-012.

10CFR50.59 Evaluation Summary:

This design change was performed to simplify the procedures for bypassing the EOP support procedures identified above. The bypasses were accomplished using knife switches located on control room panels C-03, C-04, and C-25. The Technical Specifications did require revision as a result of this design change. The EOPs were revised to include the function and usage of the revised bypass circuit. The knife switches did not create an unreviewed safety question as determined by 10CFR50.59 evaluation.

17. MET Wind Sensor Upgrade (Design Change 99Q205)

Description:

The MET wind speed and direction sensors on the primary and backup towers were replaced with new combination sensors.

10CFR50.59 Evaluation Summary:

The replaced wind speed and direction sensors were susceptible to damage by high winds and blowing debris. The new combination sensors are less susceptible to damage and require minimal maintenance and calibration.

18. Convert LS 3063 to Bubbler System (Design Change 00Q135)

Description:

This modification changed the method by which the level switch LS-3063, "Turbine Building Floor Drain Hi Level Alarm", initiating condition is sent to the switch to actuate the high level annunciator. Previously an air capture tube was used. This was converted to an air bubbler system. The air bubbler system was installed under a temporary modification (Jumper/Bypass). This modification allowed the Jumper/Bypass to be removed.

10CFR50.59 Evaluation Summary:

This modification changed the method by which level switch LS-3063 initiating condition is sent to the switch to actuate the high level annunciator. Previously an air capture tube was used. It was converted to an air bubbler system. This modification was non-safety related, non-QA related, non-security related, non-fire related, and did not change the Monticello Technical Specifications. The P&ID for air system and radwaste in Chapter 15 of the USAR required revision. Addition of the new tubing and supports described in the modification did not create an unreviewed safety question.

19. Fuel Zone Level Instrumentation Modification (01Q075)

Description:

The purpose of design change 01Q075 was to improve the reliability of the reactor fuel zone level instruments (LT-2-3-112A and LT-2-3-112B) during accident conditions by changing the instrument reference leg sensing line from the feedwater reference columns to the safeguards columns. In addition to the fuel zone level instruments, several other instruments, such as the reactor low low set pressure transmitters and the ECCS 2/3 core height interlock level switches, were rerouted to the safeguards column. The utilization of the safeguards column for these instruments minimized the potential for these instruments to become inoperable due to the reference leg flashing to steam during accident conditions.

10CFR50.59 Evaluation Summary:

In summary, the purpose of design change 01Q075 was to improve the reliability of the reactor fuel zone level instruments (LT-2-3-112A and LT-2-3-112B) during accident conditions by changing the instrument reference leg sensing line from the feedwater reference columns to the safeguards columns. In addition to the fuel zone level instruments, several other instruments, such as the reactor low low set pressure transmitters and the ECCS 2/3 core height interlock level switches, were rerouted to the safeguards column. The utilization of the safeguards column for these instruments minimized the potential for these instruments to become inoperable due to the reference leg flashing to steam during accident conditions. It was determined that an unreviewed safety question did not exist.

20. Remove Spool Piece from Solid Radwaste System and Install Air Lance Insertion Point (Design Change 00Q325)

Description:

To achieve proper mixing in condensate phase separator tanks T-34A and T-34B, an air lance was inserted through the manhole at the top of the tank to the bottom of the tank and service air was admitted through the air lance. This resulted in a large gap between the manhole cover and the upper rim of the manhole that allowed resin to spray upwards and out of the gap. This modification allowed the insertion of the air lance without having to remove the manhole cover by cutting a hole in the manhole cover and affixing a thick flexible rubber gasket to the hole. A blank flange is installed over the hole when not in use.

10CFR50.59 Evaluation Summary:

Previously the method to achieve proper mixing in condensate phase separator tanks T-34A and T-34B was to insert an air lance through the manhole at the top of the tank to the bottom of the tank and admit plant service air through this air lance. This resulted in a large gap between the manhole cover and the upper rim of the manhole that allowed resin to spray upwards and out of the gap. This modification allowed the insertion of the air lance without having to remove the manhole cover by cutting a hole in the manhole cover and affixing a thick flexible rubber gasket to the hole. The hole is just large enough to pass the air lance through the gasket, thus eliminating the need to remove the manhole cover. This eliminated the large gap between the manhole cover and the upper rim of the manhole and prevented resin and sludge from spraying out of the tank. A blank flange is installed over the hole when not in use. Removal of the spool piece and installing blank flanges on line RWN51-4-HC, as described in the scope of this modification, did not create an unreviewed safety question.

21. Chilled Water Vent Valves and V-CC-10 Bypass (Design Change 98Q170)

Description:

This project modified the steam chase supply cooling coil (V-CC-10) configuration to aid in the monitoring of cooling coil performance and to prevent further performance degradation. This project also added vent valves and piping to the chilled water system to aid system start-up and lay-up evolutions.

10CFR50.59 Evaluation Summary:

To reduce the potential for performance degradation of cooling coil V-CC-10 and the resulting increase in steam chase ambient temperature, the following equipment was installed by this design change: filter upstream of the coil, bypass ductwork and associated isolation damper, dP gage across the coil and filter, temperature indicator, and an inspection platform. Vent piping to aid start up and shutdown of the chilled water system was also added. This modification did not create an unreviewed safety question.

22. Full Steam Dilution Recombiner (Design Change 99Q160)

Description:

Modification 99Q160, Full Steam Dilution Recombiner, improves the operational safety of the recombiner system. The potential for a hydrogen burn or detonation within the system is essentially eliminated by diluting the stoichiometric mixture of hydrogen and oxygen with steam. Steam dilution is accomplished by converting the second stage Steam Jet Air Ejectors (SJAEs) to a non-condensing stage; second stage motive steam remains in the process to dilute the offgas mixture.

10CFR50.59 Evaluation Summary:

Modification 99Q160, Full Steam Dilution Recombiner, improved the operational safety of the recombiner system. The potential for a hydrogen burn or detonation within the system was essentially eliminated by diluting the stoichiometric mixture of hydrogen and oxygen with steam. A 10CFR50.59 evaluation concluded that no unreviewed safety question was created by the modification.

23. RHRSW Motor Cooling Coil Pipe Coupling (Design Change 00Q220)

Description:

The cooling water piping was shortened to allow for the installation of a flex hose between the piping and the cooling water connection at the motor.

10CFR50.59 Evaluation Summary:

The RHR Service Water (RHRSW) cooling coil supply and discharge piping for RHRSW pumps was modified to allow for geometry differences at the cooling coil connections on the spare and inservice pump motors. The cooling water piping was shortened to allow for the installation of a flex hose between the piping and the cooling water connection at the motor. A 10CFR50.59 evaluation concluded that this design change did not present an unreviewed safety question.

24. St. Cloud 115KV Line Designation Change (Mod 00Q330)

Description:

This design change revised site drawings, documents, and the USAR to reflect the 115KV transmission line name change from St. Cloud to Industrial Park.

10CFR50.59 Evaluation Summary:

This design change does not represent an unreviewed safety question. There were no Technical Specification changes. This design change revised site drawings, documents, and the USAR to reflect the 115KV transmission line name change from St. Cloud to Industrial Park.

25. SBGT Makeup Air Improvements (Design Change 00Q270)

Description:

The standby gas treatment (SBGT) system provides, whenever secondary containment isolation conditions exist, a small negative pressure to minimize ground level escape of airborne radioactivity. Filters are provided in the system to remove radioactive particulates, and charcoal adsorbers are provided to remove radioactive halogens. All flow from the SBGT System is released through the offgas vent stack and continuously monitored by the stack gas monitoring system.

The objective of this design change was to minimize filtration bypass. Several items were accomplished to meet this goal, including: (1) removal of the abandoned mixing box, associated ducting, back draft damper, and air supply pressure regulator; (2) blanking of the abandoned outside air makeup duct; (3) addition of additional makeup air capacity from the turbine building.

10CFR50.59 Evaluation Summary:

This design change prevented potential standby gas treatment filtration bypass by increasing the flow of makeup air into the room by removing the mixing air box and associated equipment and increasing the vent area to the room from the turbine building. This design change did not represent an unreviewed safety.

26. Single Loop Recirc Operation with the Idle Loop Discharge Valve Closed (SRI 01-002)

Description:

From time to time it is desired to operate the Monticello Plant with one recirculation loop out of service ("single loop operation"). Previously this was done with the pump discharge valve in the shutdown loop essentially closed. This led to difficulties because the Technical Specifications contain temperature difference limits that preclude pump restart if these limits are exceeded. With the discharge valve essentially closed, loop cooldown resulted in potentially exceeding these limits when the loop is out of service for more than an hour or so. This loop cooldown would, in turn, require the reactor to be shut down and depressurized to a significant extent to allow the idle pump to be restarted, and could result in a reactivity transient. The severity of this transient would depend on the specific conditions at the time.

It is expected that opening the recirculation pump discharge valve on the idle loop and sufficiently increasing the speed of the operating pump will result in sufficient water circulation through the idle loop to prevent significant loop cooldown, thereby allowing restart of the idle loop when desired without having to depressurize the reactor.

The recirculation system does not provide a safety-related function other than that it serves as primary system boundary and as a Low Pressure Coolant Injection (LPCI) injection flow path. The pump discharge valve must close when required by the LPCI system logic.

The purpose of this SRI was to provide justification for fully opening the pump discharge valve in an idle recirculation loop. This required a USAR change because USAR Section 4.3.2.1 makes the statement, "In the event that one pump fails or is shut off, the discharge valve in the inoperative driving loop would be manually closed," with no further qualification or information.

10CFR50.59 Evaluation Summary:

SRI 01-002 has shown that the position of the discharge valve of the idle recirculation pump does not affect the acceptability of single loop operation.

EXHIBIT B

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO LICENSEE DOCKETED COMMITMENTS

The purpose of this exhibit is to provide a brief description and a summary of changes to formally tracked commitments established with the NRC by the Monticello Nuclear Generating Plant. These commitments are being identified and reported to the Commission in accordance with guidance provided in NEI technical report 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes."

1. Monticello Commitment M90001A

Source Document: Monticello Licensee Event Report 89-040, "Failure to Meet Secondary Containment Performance Requirements Due to Design Deficiencies"

Commitment: Place administrative hold on AO-2982 to secure it in closed position when SBGT is required to be operable.

Change: Operating procedures to control AO-2982 in the closed position when SBGT is required to be operable. The commitment does not say a hold tag needs to be placed on the switch, only that an administrative hold must secure AO-2982 in the closed position during SBGT operation. The permanent removal of the hold tags allows Operations to control AO-2982 by procedures and enter LCOs as necessary.

2. Monticello Commitr '84119A

Source Document: Technical Evaluation Report TER-C5506-370– Control of Heavy Loads, January 30, 1984

Commitment: Loads of weight greater than one fuel element (excluding the crane load blocks and associated tackle) shall not be transported directly over spent fuel stored in the spent fuel pool without prior NRC approval.

Change: Loads of weight greater than the weight assumed in the refueling accident analysis for one fuel element and its associated grapple assembly shall not be transported directly over spent fuel stored in the spent fuel pool without prior NRC approval. This commitment is contained in the NRC Safety Evaluation Report which accepted NSP's response to NUREG-0612, Control of Heavy Loads at Nuclear Power Plants – Resolution of Generic Technical Activity A-36, July 1980. The revised commitment meets the intent of the original commitment. The weight limit described in the revised commitment as the weight of one fuel element and its associated grapple assembly is consistent with the heavy load limit over the spent fuel pool established in accordance with NUREG-0612 as discussed in Section 12 of the USAR (Revision 18). The heavy load limit over the spent fuel pool is any load greater than weight of one fuel element and its associated grapple assembly. The revised commitment is also consistent with the refueling accident analysis discussed in Section 14 of the USAR (Revision 18) which is based on the weight of one fuel element and its associated grapple assembly. The refueling accident analysis shows that the activity release due to a drop of a fuel element and its associated grapple assembly onto the reactor core of the spent fuel stored in the spent fuel pool is well below the 10CFR100 limits.

EXHIBIT C

MONTICELLO NUCLEAR GENERATING PLANT
SUMMARY OF INFORMATION REMOVED FROM THE USAR

Consistent with the guidance in Nuclear Energy Institute (NEI) Technical Report 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, and Regulatory Guide 1.181, information removed from the Monticello USAR is summarized below.

- A USAR statement that the High Pressure Coolant Injection (HPCI) pump discharge valves are prevented from opening automatically whenever a turbine trip condition exists was incorrect. These valves are not required to close in support of any safety- related function or operational requirement under HPCI turbine trip conditions. The statement was deleted from Section 6.2.4.2.4
- Section 7.11.2 and Table 7.11-1 are to be deleted because information in these sections was obsolete. The discussion did not reflect the current Loss of Coolant Accident (LOCA) analyses, nor did it reflect the current plant abnormal operating procedures.
- In Section 3.3.3.1, references to total peaking factor were eliminated. Monticello has phased out the use of this term since it is redundant to the use of linear heat generation rate (LHGR).
- Deleted ultrasonic testing (UT) inspection results from Section 3.6.2.1 because the material presented was outdated (from 1994) and did not reflect the current inspection. This information also did not advance the core shroud description of which it was a part.
- The plant was initially constructed with a combined iodine/particulate airborne activity monitor (CIM/CAM sampler). In 1978, this unit was replaced with the present Drywell CAM Particulate Monitor; the iodine channel was removed. References to the CIM were removed from USAR Section 4.3.3.3.
- Deleted references to line D12.5-EF/EB from the High Energy Line Break (HELB) discussion in Appendix I. A plant modification resulted in a pressure reduction in this line, which removed it from the HELB category.
- Deleted the description of how the Main Steam Isolation Valve (MSIV) double disk wedge assembly functions to equalize and minimize seat wear during valve closures from Section 5.2.2.5.3 as a result of a plant alteration. The alteration was installed to address damage to the outboard MSIVs due to "wind-milling" of the valve disks.
- In Section 7.2.2.2, information made obsolete due to complete decoupling of the turbine control and the recirculation flow control systems by a past modification was deleted.
- The portion of Section 13.3.5 that discusses overtime restrictions was eliminated. Overtime restrictions are described by Technical Specification 6.1.F.
- References to the Portable Cement Solidification System referred to in Section 9.4.2.2.2 were removed. This system is not located on-site and is no longer available from the vendor (CNSI). This equipment is not used at Monticello for waste solidification.
- A reference to a vacuum breaker analysis submitted to in response Generic Letter (GL) 88-03, but not used as a basis for the NRC Safety Evaluation Report (SER) on Monticello's response to that letter, was eliminated in Section 5.2.1.2.3. This

information was extraneous because NRC approval of Monticello's response to GL 88-03 was not predicated on this analysis.

- References to responses made by the Dresden facility in response to Atomic Energy Commission (AEC) questions were deleted from Section 7.6.1.2.9.

EXHIBIT E

REPORT OF CHANGES TO MONTICELLO FIRE PROTECTION PROGRAM

This section contains a report of changes to the Monticello Fire Protection Program (FPP) in accordance with the provisions of 10 CFR 50.71(e), 10 CFR 50.59, and Generic Letter (GL) 86-10.

In conformance with GL 86-10, the Updated Fire Hazards Analysis (FHS) and Safe Shutdown Analysis (SSA) are incorporated by reference into the Updated Safety Analysis Report (USAR). These reports underwent significant revision in the course of this USAR revision cycle. Electronic copies of the revised FHA and SSA are provided on the CD-ROM accompanying this submittal.

Amendment 119 to the Monticello Facility Operating License changed license condition C.2.4 to conform to Generic Letter 86-10, and relocated the FPP from the Technical Specifications to a licensee controlled FPP. Electronic copies of the following program documents, as incorporated by reference into the USAR, are provided on the enclosed CD-ROM:

- Fire Protection Program Plan (4 AWI-08.01.00, Revision 0)
- Fire Prevention Practices (4 AWI-08.01.01, Revision 19)

Limiting Conditions for Operation and Surveillance Requirements for Fire Detection and Protection Systems are now defined as Impairments. These were relocated from the Technical Specifications as a part of the aforementioned License Amendment to a site implementing procedure. No changes were made to the Impairments subsequent to their transfer to the implementing procedure.

Consistent with the requirements of the Monticello FPP, a summary of occasions on which more than one fire pump is simultaneously inoperable is to be provided to the NRC with the summary of program changes. On two occasions, three fire pumps were declared inoperable for performance of flow capability testing. During this surveillance, two out of three pumps are considered inoperable when their respective hand-switches are turned to the OFF/STOP position to prevent the pumps from automatically starting when testing the third pump. On both occasions, performance of the flow capability test was completed and the pumps were declared operable in approximately four hours. If the fire pumps had been required during the performance of flow testing, provisions were in place to abort the surveillance and return the pumps to service.

LIST OF EFFECTIVE PAGES

Section	Revision
VOLUME 1	
List of Effective Pages	19
T of C Section	19
1 TOC	19
1.1	19
1.2	19
1.3	19
1.4	19
1.5	19
1.FIGURES	19
2 TOC	19
2.1	19
2.2	19
2.3	19
2.4	19
2.5	19
2.6	19
2.7	19
2.8	19
2.9	19
2.10	19
2.FIGURES	19
3 TOC	19
3.1	19
3.2	19
3.3	19
3.4	19
3.5	19
3.6	19
3.7	19
3.FIGURES	19
4 TOC	19
4.1	19
4.2	19
4.3	19
4.4	19
4.5	19
4.6	19
4.7	19
4.8	19
4.FIGURES	19

LIST OF EFFECTIVE PAGES

Section	Revision
VOLUME 2	
5 TOC	19
5.1	19
5.2	19
5.3	19
5.4	19
5.FIGURES	19
6 TOC	19
6.1	19
6.2	19
6.3	19
6.4	19
6.5	19
6.6	19
6.7	19
6.8	19
6.FIGURES	19
7 TOC	19
7.1	19
7.2	19
7.3	19
7.4	19
7.5	19
7.6	19
7.7	19
7.8	19
7.9	19
7.10	19
7.11	19
7.12	19
7.13	19
7.14	19
7.FIGURES	19
VOLUME 3	
8 TOC	19
8.1	19
8.2	19
8.3	19
8.4	19
8.5	19
8.6	19
8.7	19
8.8	19
8.9	19
8.10	19
8.11	19

LIST OF EFFECTIVE PAGES

Section	Revision
8.12	19
8.13	19
8.FIGURES	19
9 TOC	19
9.1	19
9.2	19
9.3	19
9.4	19
9.5	19
9.FIGURES	19
10 TOC	19
10.1	19
10.2	19
10.3	19
10.4	19
10.5	19
10.FIGURES	19
11 TOC	19
11.1	19
11.2	19
11.3	19
11.4	19
11.5	19
11.6	19
11.7	19
11.8	19
11.FIGURES	19
VOLUME 4	
12 TOC	19
12.1	19
12.2	19
12.3	19
12.4	19
12.5	19
12.FIGURES	19
13 TOC	19
13.1	19
13.2	19
13.3	19
13.4	19
13.5	19
13.6	19
13.7	19
13.8	19
13.FIGURES	19

LIST OF EFFECTIVE PAGES

Section	Revision
14 TOC	19
14.1	19
14.2	19
14.3	19
14.4	19
14.5	19
14.6	19
14.7	19
14.8	19
14.9	19
14.10	19
14.11	19
14.FIGURES	19
APPENDIX 14A	NAD-MN-004, Revision 0

VOLUME 5

Section	Revision
15 TOC	19

Drawing Number	Revision
NX-7834-2-2	C
NX-7865-7-1	B
NX-7865-7-2	1
NX-7993-1-1	B
NH-36032 (M-100)	N
NH-36034 (M-103)	AF
NH-36035 (M-104)	AJ
NH-36035-2 (M-104-1)	N
NH-36036 (M-105)	AQ
NH-36037 (M-106)	AP
NH-36038 (M-107)	AA
NH-36038-2 (M-107)	R
NH-36041 (M-110)	BP
NH-36041-2 (M-110-2)	F
NH-36042 (M-111)	AD
NH-36042-2 (M-111-1)	E
NH-36043 (M-137)	AA
NH-36044 (M-138)	AP
NH-36045 (M-139)	W
NH-36046 (M-140)	X
NH-36046-2 (M-140)	D
NH-36047-1 (M-141)	AC
NH-36047-2 (M-141-2)	J
NH-36049-1 (M-131)	AW
NH-36049-2 (M-131)	AD
NH-36049-3 (M-131)	AN
NH-36049-4 (M-131)	T
NH-36049-11 (M-131)	N
NH-36049-12 (M-131)	Q

Drawing Number	Revision
NH-36049-14 (M-131)	AA
NH-36049-15 (M-131)	A
NF-36054 (M-1)	F
NF-36055 (M-2)	Q
NF-36055-1	E
NF-36056 (M-3)	M
NF-36056-1	H
NF-36057 (M-4)	H
NF-36058 (M-5)	C
NF-36059 (M-6)	Q
NF-36060 (M-7)	U
NF-36061 (M-8)	H
NF-36062 (M-9)	C
NF-36063 (M-10)	A
NH-36159 (M-142)	AB
NH-36159-2 (M-142)	E
NF-36175 (E-1)	AE
NF-36222 (C-3)	G
NF-36223 (C-4)	E
NH-36241 (M-115)	AY
NH-36242 (M-116)	AW
NH-36242-1 (M-116-1)	D
NH-36242-2 (M-116-2)	A
NH-36243 (M-117)	AL
NH-36243-1 (M-117)	J
NH-36244 (M-118)	AW
NH-36245 (M-119)	S
NH-36246 (M-120)	BH
NH-36247 (M-121)	BL
NH-36248 (M-122)	AH
NH-36249 (M-123)	AJ
NH-36250 (M-124)	AA
NH-36251 (M-125)	AM
NH-36252 (M-126)	AA
NH-36253 (M-127)	V
NH-36254 (M-128)	AU
NH-36255 (M-129)	AB
NH-36256 (M-135)	AB
NH-36257 (M-136)	S
NH-36258 (M-143)	AY
NH-36263 (M-151)	L
NH-36266 (M-154)	K
NH-36489 (M-800)	AC
NH-36489-2 (M-800-1)	L
NH-36516 (M-810)	Y
NE-36640-2 (E-110-2)	J
NE-36640-3 (E-110-3)	AA
NE-36640-4 (E-110-4)	J
NE-36640-4-2 (E-110-4B)	E
NE-36640-6 (E-110-6)	G
NH-36664 (M-112)	BG

Drawing Number	Revision
NH-36665 (M-811)	CE
NH-36666 (M-812)	AT
NE-36771-4 (E-111-4)	G
NH-36881 (M-145)	S
NH-46162 (M-130)	AV
NH-54817-4	N
NF-54822-2	E
NF-54823-2	G
NF-54823-3	C
NF-54823-4	F
NH-91197	K
NE-93523-3 (E-201-3)	F
NE-93523-4 (E-201-4)	E
NH-94896	N
NH-94897	M
ND-95208	E
ND-95209	S
NH-119062	A
NH-178635	B

VOLUME 6

Page	Revision	Page	Revision
APPENDIX A Tab	NA	A.1-18	4
A-i	4	A.2-i	4
A-ii	4	A.2-ii	4
A-iii	13	A.2-iii	4
A.1-i	4	A.2-iv	4
A.1-ii	4	A.2-1	4
A.1-iii	4	A.2-2	4
A.1-iv	4	A.2-3	4
A.1-v	4	A.2-4	4
A.1-1	4	A.2-5	4
A.1-2	4	A.2-6	4
A.1-3	4	A.2-7	4
A.1-4	4	A.2-8	4
A.1-5	4	A.2-9	4
A.1-6	4	A.2-10	4
A.1-7	4	A.2-11	4
A.1-8	4	A.2-12	4
A.1-9	4	A.2-13	4
A.1-10	4	A.2-14	4
A.1-11	4	A.2-15	13
A.1-12	4	A.2-16	4
A.1-13	4	A.2-17	4
A.1-14	4	A.2-18	18
A.1-15	4	A.2-19	4
A.1-16	4	A.2-20	4
A.1-17	4	A.2-21	4

Page	Revision
A.2-22	4
A.2-23	4
A.2-24	4
A.2-25	4
A.2-26	4
A.2-27	4
A.2-28	4
A.2-29	4
A.2-30	4
A.2-31	4
A.2-32	4
A.2-33	4
A.2-34	4
A.2-35	4
A.2-36	4
A.2-37	4
A.2-38	4
A.2-39	4
A.2-40	4
A.2-41	4
A.2-42	4
A.2-43	4
A.2-44	4
A.2-45	4
A.2-46	4
A.2-47	4
A.2-48	4
A.2-49	4
A.2-50	4
A.2-51	4
A.2-52	4
A.2-53	4
A.2-54	4
A.2-55	4
A.2-56	4
A.2-57	4
A.2-58	4
A.2-59	4
A.2-60	4
A.3-i	4
A.3-ii	4
A.3-ii	4
A.3-iv	4
A.3-1	4
A.3-2	4
A.3-3	4
A.3-4	4
A.3-5	4
A.3-6	4
A.3-7	4
A.3-8	4

Page	Revision
A.3-9	4
A.3-10	4
A.3-11	4
A.3-12	4
A.3-13	4
A.3-14	4
A.3-15	4
A.3-16	4
A.3-17	4
A.3-18	4
A.3-19	4
A.3-20	4
A.3-21	4
A.3-22	4
A.3-23	4
A.3.24	4
A.4-i	4
A.4-ii	4
A.4-iii	4
A.4-iv	4
A.4-1	4
A.4-2	4
A.4-3	4
A.4-4	4
A.4-5	4
A.4-6	4
A.4-7	4
A.4-8	4
A.4-9	4
A.4-10	4
A.4-11	4
A.4-12	4
A.4-13	4
A.4-14	4
A.4-15	4
A.4-16	18
A.4-17	18
A.4-18	18
A.4-19	4
A.4-20	4
A.4-21	4
A.4-22	4
A.4-23	18
A.4-24	4
A.4-25	4
A.4-26	4
A.4-27	4
A.4-28	4
A.4-29	4
A.4-30	4
A.4-31	4

Page	Revision	Page	Revision
A.4-32	4	A.5-32	4
A.4-33	4	A.5-33	4
A.4-34	4	A.5-34	4
A.4-35	4	A.5-35	4
A.4-36	4	A.6-i	4
A.4-37	4	A.6-1	4
A.4-38	4	A.6-2	4
A.4-39	4	A.6-3	4
A.4-40	4	A.6-4	4
A.4-41	4	A.6-5	4
A.4-42	4	A.6-6	4
A.4-43	4	A.6-7	4
A.4-44	4	A.6-8	4
A.4-45	4	A.6-9	4
A.4-46	4	A.6-10	4
A.4-47	4	A.6-11	4
A.4-48	4	A.6-12	4
A.5-i	4	A.6-13	4
A.5-ii	4	A.6-14	4
A.5-iii	4	A.6-15	4
A.5-1	4	A.6-16	4
A.5-2	4	A.6-17	4
A.5-3	4	A.6-18	4
A.5-4	4	A.6-19	4
A.5-5	4	A.6-20	4
A.5-6	4	A.6-21	4
A.5-7	4	A.6-22	4
A.5-8	4	A.6-23	4
A.5-9	4	A.7-i	4
A.5-10	4	A.7-1	4
A.5-11	4	A.7-2	4
A.5-12	4	A.7-3	4
A.5-13	4	A.7-4	4
A.5-14	4	A.7-5	4
A.5-15	4	A.7-6	4
A.5-16	4	A.7-7	4
A.5-17	4	A.7-8	4
A.5-18	4	A.7-9	4
A.5-19	4	A.7-10	4
A.5-20	4	A.7-11	4
A.5-21	4	A.8-i	4
A.5-22	4	A.8-1	4
A.5-23	4	A.8-2	4
A.5-24	4	A.8-3	4
A.5-25	4	A.8-4	4
A.5-26	4	A.8-5	4
A.5-27	4	A.8-6	4
A.5-28	4	A.9-i	4
A.5-29	4	A.9-ii	4
A.5-30	4	A.9-iii	13
A.5-31	4	A.9-1	4

Page	Revision
A.9-2	4
A.9-3	4
A.9-4	13
A.9-5	4
A.9-6	4
A.9-7	4
A.9-8	4
A.9-9	4
A.9-10	4

- VOLUME 7 -

APPENDIX C Tab	NA
Operational QA Plan Rev 24	
Cover Sheet	
Table of Contents	
Pages 1 through 102	
APPENDIX D Tab	NA
D TOC	19
D.1	19
D.2	19
D.3	19
D.4	19
D.5	19
APPENDIX E Tab	NA
E TOC	19
E.1	19
E.2	19
APPENDIX F Tab	NA
F-i	18
F-ii	4
F.1-1	4
F.1-2	4
F.2-1	4
F.2-2	4
F.2-3	4
F.2-4	4
F.2-5	4
F.2-6	4
F.2-7	4
F.2-8	18
FIG F.2.1	4
FIG F.2.2	4
F.3-1	4

Page	Revision
F.4-1	4
F.4.2	4
F.A-i	4
F.A-1	4
F.A-2	4
F.A-3	4
F.A-4	4
F.A-5	4
F.A-6	4
F.A-8	4
F.A-9	4
F.A-10	4
F.A-11	4
F.A-12	4
F.A-13	4
F.A-14	4
F.A.15	4
F.A.A-i	4
F.A.A-1	4
F.A.B-i	4
F.A.B-1	4
F.A.B-2	4
F.A.B-3	4
F.A.B-4	4
F.A.B-5	4
F.A.C-i	4
F.A.C-1	4
F.A.D-i	4
F.A.D-1	4
F.A.D-2	4
F.A.D-3	4
F.A.E-i	4
F.A.E-1	4
F.A.E-2	4
F.A.E-3	4
F.A.E-4	4
F.A.E-5	4
F.A.E-6	4
F.A.F-i	4
F.A.F-1	4
F.A.F-2	4
F.A.F-3	4
F.A.F-4	4
F.A.F-5	4
F.A.F-6	4
F.A.F-7	4
F.A.F-8	4
F.A.F-9	4
F.A.F-10	4
F.A.F-11	4
F.B-i	4

Page	Revision
F.B-1	4
F.B-2	4
F.B-3	4
F.B-4	4
F.B-5	4
F.B-6	4

VOLUME 8

APPENDIX G Tab	NA
G-i	4
G-ii	4
G-iii	4
G-iv	4
G.1-1	4
G.1-2	4
G.1-3	4
G.2-1	4
G.2-2	4
G.2-3	11
G.2-4	4
G.2-5	4
G.2-6	13
G.3-1	4
G.3-2	4
G.3-3	4
G.3-4	4
G.3-5	4
G.3-6	4
G.3-7	4
G.3-8	4
G.3-9	4
G.4-1	4
G.4-2	4
G.4-3	4
G.4-4	13
G.4-5	4
G.5-1	4
EX 1	4
EX 2	4
EX 3	4
EX 4	4
EX 5	4
EX 6	4
EX 7	4
EX 8	4
EX 9	4

Page	Revision
APPENDIX H Tab	NA
H-i	4
H-ii	5
H.1-1	4
H.1-2	7
H.1-3	7
H.1-4 (blank)	7
Exhibit 1	
1-1	4
Specification 21A112, Rev. 6 pages 1-2 through 1-36	
Attachment A to Specification 21A112, Rev. 3, pages 1-37	
Attachment B to Specification 21A112, Rev. 3, pages 1-38 through 1-43	
Attachment C to Specification 21A112, Rev. 0, pages 1-44 through 1-49	
Attachment D to Specification 21A112, Rev. 1, pages 1-50 through 1-52	
Attachment E to Specification 21A112, Rev. 1 pages 1-53	
Exhibit 2	
2-1	4
2-2	4
2-3	4
2-4	4
2-5a	4
2-5b	4
2-6	4
2-7	4
2-8	4
2-9	4
2-10	4
2-11	4
2-12	4
2-13	4
2-14	4
2-15	4
2-16	4
2-17	4
2-18	4
2-19	4
2-20	4

Page	Revision
2-21	4
2-22	4
2-23	4
2-24	4
2-25	4
2-26	4
2-27	4
2-28	4
2-29	4
2-30	4
2-31	4
2-32	4
Exhibit 3	
3-1	4
3-2	4
3-3	4
3-4	4
3-5	4
3-6	4
3-7	4
3-8	4
3-9	4
3-10	4
3-11	4
3-12	4
3-13	4
3-14	4
3-15	4
3-16	4
3-17	4
3-18	4
3-19	4
3-20	4
3-21	4
3-22	4
3-23	4
3-24	4
3-25	4
3-26	4
3-27	4
3-28	4
3-29	4
3-30	4
3-31	4
3-32	4
Exhibit 4	
4-1	4
4-2	4
4-3	4
4-4	4

Page	Revision
4-5	4
4-6	4
4-7	4
4-8	4
4-9	4
4-10	4
4-11	4
4-12	4
4-13	4
4-14	4
4-15	4
4-16	4
4-17	4
4-18	4
4-19	4
4-20	4
4-21	4
4-22	4
4-23	4
4-24	4
4-25	4
4-26	4
4-27	4
4-28	4
4-29	4
4-30	4
4-31	4
4-32	4
4-33	4
4-34	4
4-35	4
4-36	4
Exhibit 5	
5-1	4
5-2	4
5-3	4
5-4	4
5-5	4
5-6	4
5-7	4
5-8	4
5-9	4
5-10	4
5-11	4
5-12	4
5-13	4
5-14	4
5-15	4
5-16	4
5-17	4

Exhibit 6	
6-1	4
6-2	4
6-3	4
Exhibit 7	
7-1	5
Specification 22A5541, Rev. 2, pages 7-2 through 7-19	
Exhibit 8	
8-1	5
Specification 22A7454, Rev. 1, pages 8-2 through 8-189	
Exhibit 9	
9-1	5
Specification 22A7227, Rev. 0, pages 9-2 through 9-18	

APPENDIX I Tab	NA
I TOC	19
I.1	19
I.2	19
I.3	19
I.4	19
I.5	19
I.6	19
I.FIGURES	19

Revision (Rev) No.	Effective Date	Date of Issue	Remarks
ORIGINAL	10/81	5/1/82	
1	10/82	4/29/83	
2	10/83	4/30/84	
3	12/84	6/28/85	
4	12/85	6/30/86	Added Appendices A thru I
5	12/86	6/30/87	
6	12/87	6/30/88	
7	12/88	6/30/89	
8	11/89	11/4/89	Modified Containment Isolation Valve List to Coordinate Removal of the List from the Tech Spec
9	12/89	6/29/90	
10	12/90	6/28/91	
11	12/91	6/30/92	
12	9/93	9/1/93	Reprint of USAR sections 1 through 14 due to placement of these sections on the Monticello Site Technical Publishing System
13	5/95	4/20/95	Reprint of USAR Appendix D, E, and I due to placement of these sections on the Monticello Site Technical Publishing System
14	5/96	11/18/96	
15	7/97	10/31/97	Revision to Section 5.2 response to Treatment of Commitments as License Conditions (TAC No. 97781) and Tech Spec Amendment 98 license condition.
16	7/98	10/23/98	General revision to reflect periodic update and initial input from USAR Review Project.
17	8/99	8/25/99	Revise sections affected by the implementation of the power rerate approved by License Amendment 102.
18	5/00	8/25/00	General revision to reflect periodic update and continuing input from USAR Review Project.

Revision (Rev) No.	Effective Date	Date of Issue	Remarks
19	6/02		General revision to reflect periodic update and continuing input from USAR Review Project.

Prepared by M Miszczak

Approved by Site Vice President or Designee

for

Approved by D Neve

I/jmr

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ______________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-01.01	19	INTRODUCTION AND SUMMARY - PURPOSE, SCOPE AND ORGANIZATION OF REPORT
9703	USAR-01.02	19	INTRODUCTION AND SUMMARY - PRINCIPAL DESIGN CRITERIA
9703	USAR-01.03	19	INTRODUCTION AND SUMMARY - SUMMARY DESIGN DESCRIPTION AND SAFETY ANALYSIS
9703	USAR-01.04	19	INTRODUCTION AND SUMMARY - IDENTIFICATION AND QUALIFICATION OF CONTRACTORS
9703	USAR-01.05	19	INTRODUCTION AND SUMMARY - REFERENCES
9703	USAR-01.FIG	19	INTRODUCTION AND SUMMARY - FIGURES
9703	USAR-01.TOC	19	INTRODUCTION AND SUMMARY - TABLE OF CONTENTS
9703	USAR-02.01	19	SITE AND ENVIRONS - INTRODUCTION
9703	USAR-02.02	19	SITE AND ENVIRONS - SITE DESCRIPTION
9703	USAR-02.03	19	SITE AND ENVIRONS - METEOROLOGY
9703	USAR-02.04	19	SITE AND ENVIRONS - HYDROLOGY
9703	USAR-02.05	19	SITE AND ENVIRONS - GEOLOGY AND SOIL INVESTIGATION
9703	USAR-02.06	19	SITE AND ENVIRONS - SEISMOLOGY
9703	USAR-02.07	19	SITE AND ENVIRONS - RADIATION ENVIRONMENTAL MONITORING PROGRAM (REMP)
9703	USAR-02.08	19	SITE AND ENVIRONS - ECOLOGICAL AND BIOLOGICAL STUDIES
9703	USAR-02.09	19	SITE AND ENVIRONS - CONSEQUENCES OF HYPOTHETICAL LOCAL CATASTROPHES
9703	USAR-02.10	19	SITE AND ENVIRONS - REFERENCES
9703	USAR-02.FIG	19	SITE AND ENVIRONS - FIGURES
9703	USAR-02.TOC	19	SITE AND ENVIRONS - TABLE OF CONTENTS
9703	USAR-03.01	19	REACTOR - GENERAL SUMMARY
9703	USAR-03.02	19	REACTOR - THERMAL AND HYDRAULIC CHARACTERISTICS
9703	USAR-03.03	19	REACTOR - NUCLEAR CHARACTERISTICS
9703	USAR-03.04	19	REACTOR - FUEL MECHANICAL CHARACTERISTICS
9703	USAR-03.05	19	REACTOR - REACTIVITY CONTROL MECHANICAL CHARACTERISTICS
9703	USAR-03.06	19	REACTOR - OTHER REACTOR VESSEL INTERNALS
9703	USAR-03.07	19	REACTOR - REFERENCES
9703	USAR-03.FIG	19	REACTOR - FIGURES
9703	USAR-03.TOC	19	REACTOR - TABLE OF CONTENTS
9703	USAR-04.01	19	REACTOR COOLANT SYSTEM - SUMMARY DESCRIPTION
9703	USAR-04.02	19	REACTOR COOLANT SYSTEM - REACTOR VESSEL
9703	USAR-04.03	19	REACTOR COOLANT SYSTEM - RECIRCULATION SYSTEM

A053

Northern States Power Company
Monticello Nuclear Generating Plant
Document Control Distribution System

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ______________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-04.04	19	REACTOR COOLANT SYSTEM - REACTOR PRESSURE RELIEF SYSTEM
9703	USAR-04.05	19	REACTOR COOLANT SYSTEM - REACTOR COOLANT SYSTEM VENTS
9703	USAR-04.06	19	REACTOR COOLANT SYSTEM - HYDROGEN WATER CHEMISTRY
9703	USAR-04.07	19	REACTOR COOLANT SYSTEM - ZINC WATER CHEMISTRY (GEZIP)
9703	USAR-04.08	19	REACTOR COOLANT SYSTEM - REFERENCES
9703	USAR-04.FIG	19	REACTOR COOLANT SYSTEM - FIGURES
9703	USAR-04.TOC	19	REACTOR COOLANT SYSTEM - TABLE OF CONTENTS
9703	USAR-05.01	19	CONTAINMENT SYSTEM - SUMMARY DESCRIPTION
9703	USAR-05.02	19	CONTAINMENT SYSTEM - PRIMARY CONTAINMENT SYSTEM
9703	USAR-05.03	19	CONTAINMENT SYSTEM - SECONDARY CONTAINMENT SYSTEM AND REACTOR BUILDING
9703	USAR-05.04	19	CONTAINMENT SYSTEM - REFERENCES
9703	USAR-05.FIG	19	CONTAINMENT SYSTEM - FIGURES
9703	USAR-05.TOC	19	CONTAINMENT SYSTEM - TABLE OF CONTENTS
9703	USAR-06.01	19	PLANT ENGINEERED SAFEGUARDS - SUMMARY DESCRIPTION
9703	USAR-06.02	19	PLANT ENGINEERED SAFEGUARDS - EMERGENCY CORE COOLING SYSTEM (ECCS)
9703	USAR-06.03	19	PLANT ENGINEERED SAFEGUARDS - MAIN STEAM LINE FLOW RESTRICTIONS
9703	USAR-06.04	19	PLANT ENGINEERED SAFEGUARDS - CONTROL ROD VELOCITY LIMITERS
9703	USAR-06.05	19	PLANT ENGINEERED SAFEGUARDS - CONTROL ROD DRIVE HOUSING SUPPORTS
9703	USAR-06.06	19	PLANT ENGINEERED SAFEGUARDS - STANDBY LIQUID CONTROL SYSTEM
9703	USAR-06.07	19	PLANT ENGINEERED SAFEGUARDS - MAIN CONTROL ROOM, EFT BUILDING & TSC HABITABILITY
9703	USAR-06.08	19	PLANT ENGINEERED SAFEGUARDS - REFERENCES
9703	USAR-06.FIG	19	PLANT ENGINEERED SAFEGUARDS - FIGURES
9703	USAR-06.TOC	19	PLANT ENGINEERED SAFEGUARDS - TABLE OF CONTENTS
9703	USAR-07.01	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - SUMMARY DESCRIPTION
9703	USAR-07.02	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - REACTOR CONTROL SYSTEMS
9703	USAR-07.03	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - NUCLEAR INSTRUMENTATION SYSTEM

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ____________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-07.04	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - REACTOR VESSEL INSTRUMENTATION
9703	USAR-07.05	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - PLANT RADIATION MONITORING SYSTEMS
9703	USAR-07.06	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - PLANT PROTECTION SYSTEM
9703	USAR-07.07	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - TURBINE-GENERATOR SYSTEM INSTRUMENTATION AND
9703	USAR-07.08	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - NUMAC ROD WORTH MINIMIZER & PLANT PROCESS COMPUTER
9703	USAR-07.09	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - OTHER SYSTEMS CONTROL AND INSTRUMENTATION
9703	USAR-07.10	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - SEISMIC AND TRANSIENT PERFORMANCE INSTRUMENTATION
9703	USAR-07.12	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - DETAILED CONTROL ROOM DESIGN REVIEW
9703	USAR-07.13	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - SAFETY PARAMETER DISPLAY SYSTEM
9703	USAR-07.14	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - REFERENCES
9703	USAR-07.FIG	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - FIGURES
9703	USAR-07.TOC	19	PLANT INSTRUMENTATION AND CONTROL SYSTEMS - TABLE OF CONTENTS
9703	USAR-08.01	19	PLANT ELECTRICAL SYSTEMS - SUMMARY
9703	USAR-08.02	19	PLANT ELECTRICAL SYSTEMS - TRANSMISSION SYSTEM
9703	USAR-08.03	19	PLANT ELECTRICAL SYSTEMS - AUXILIARY POWER SYSTEM
9703	USAR-08.04	19	PLANT ELECTRICAL SYSTEMS - PLANT STANDBY DIESEL GENERATOR SYSTEMS
9703	USAR-08.05	19	PLANT ELECTRICAL SYSTEMS - DC POWER SUPPLY SYSTEMS
9703	USAR-08.06	19	PLANT ELECTRICAL SYSTEMS - REACTOR PROTECTION SYSTEM POWER SUPPLIES
9703	USAR-08.07	19	PLANT ELECTRICAL SYSTEMS - INSTRUMENTATION AND CONTROL AC POWER SUPPLY SYSTEM
9703	USAR-08.08	19	PLANT ELECTRICAL SYSTEMS - ELECTRICAL DESIGN CONSIDERATIONS
9703	USAR-08.09	19	PLANT ELECTRICAL SYSTEMS - ENVIRONMENTAL QUALIFICA TION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ______________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-08.10	19	PLANT ELECTRICAL SYSTEMS - ADEQUACY OF STATION ELECTRICAL DISTRIBUTION SYSTEM VOLTAGES
9703	USAR-08.11	19	PLANT ELECTRICAL SYSTEMS - POWER OPERATED VALVES
9703	USAR-08.12	19	PLANT ELECTRICAL SYSTEMS - STATION BLACKOUT
9703	USAR-08.13	19	PLANT ELECTRICAL SYSTEMS - REFERENCES
9703	USAR-08.FIG	19	PLANT ELECTRICAL SYSTEMS - FIGURES
9703	USAR-08.TOC	19	PLANT ELECTRICAL SYSTEMS - TABLE OF CONTENTS
9703	USAR-09.01	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - SUMMARY DESCRIPTION
9703	USAR-09.02	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - LIQUID RADWASTE SYSTEM
9703	USAR-09.03	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - GASEOUS RADWASTE SYSTEM
9703	USAR-09.04	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - SOLID RADWASTE SYSTEM
9703	USAR-09.05	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - REFERENCES
9703	USAR-09.FIG	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - FIGURES
9703	USAR-09.TOC	19	PLANT RADIOACTIVE WASTE CONTROL SYSTEMS - TABLE OF CONTENTS
9703	USAR-10.01	19	PLANT AUXILIARY SYSTEMS - SUMMARY DESCRIPTION
9703	USAR-10.02	19	PLANT AUXILIARY SYSTEMS - REACTOR AUXILIARY SYSTEMS
9703	USAR-10.03	19	PLANT AUXILIARY SYSTEMS - PLANT SERVICE SYSTEMS
9703	USAR-10.04	19	PLANT AUXILIARY SYSTEMS - PLANT COOLING SYSTEM
9703	USAR-10.05	19	PLANT AUXILIARY SYSTEMS - REFERENCES
9703	USAR-10.FIG	19	PLANT AUXILIARY SYSTEMS - FIGURES
9703	USAR-10.TOC	19	PLANT AUXILIARY SYSTEMS - TABLE OF CONTENTS
9703	USAR-11.01	19	PLANT POWER CONVERSION SYSTEMS - SUMMARY DESCRIPTION
9703	USAR-11.02	19	PLANT POWER CONVERSION SYSTEMS - TURBINE-GENERATOR SYSTEM
9703	USAR-11.03	19	PLANT POWER CONVERSION SYSTEMS - MAIN CONDENSER SYSTEM
9703	USAR-11.04	19	PLANT POWER CONVERSION SYSTEMS - MAIN TURBINE BYPASS SYSTEM
9703	USAR-11.05	19	PLANT POWER CONVERSION SYSTEM - CIRCULATING WATER SYSTEM

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ______________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-11.06	19	PLANT POWER CONVERSION SYSTEMS - COOLING TOWER SYSTEM
9703	USAR-11.07	19	PLANT POWER CONVERSION SYSTEMS - CONDENSATE DEMINERALIZER SYSTEM
9703	USAR-11.08	19	PLANT POWER CONVERSION SYSTEMS - CONDENSATE AND REACTOR FEEDWATER SYSTEMS
9703	USAR-11.FIG	19	PLANT POWER CONVERSION SYSTEMS - FIGURES
9703	USAR-11.TOC	19	PLANT POWER CONVERSION SYSTEMS - TABLE OF CONTENTS
9703	USAR-12.01	19	PLANT STRUCTURES AND SHIELDING - SUMMARY DESCRIPTION
9703	USAR-12.02	19	PLANT STRUCTURES AND SHIELDING - PLANT PRINCIPAL STRUCTURES AND FOUNDATIONS
9703	USAR-12.03	19	PLANT STRUCTURES AND SHIELDING - SHIELDING AND RADIATION PROTECTION
9703	USAR-12.04	19	PLANT STRUCTURES AND SHIELDING - RADIOACTIVE MATERIALS SAFETY
9703	USAR-12.05	19	PLANT STRUCTURES AND SHIELDING - REFERENCES
9703	USAR-12.FIG	19	PLANT STRUCTURES AND SHIELDING - FIGURES
9703	USAR-12.TOC	19	PLANT STRUCTURES AND SHIELDING - TABLE OF CONTENTS
9703	USAR-13.01	19	PLANT OPERATIONS - SUMMARY DESCRIPTION
9703	USAR-13.02	19	PLANT OPERATIONS - ORGANIZATION, RESPONSIBILITIES, AND QUALIFICATIONS
9703	USAR-13.03	19	PLANT OPERATIONS - PERSONNEL EXPERIENCE AND TRAINING
9703	USAR-13.04	19	PLANT OPERATIONS - OPERATIONAL PROCEDURES
9703	USAR-13.05	19	PLANT OPERATIONS - OPERATIONAL RECORDS AND REPORTING REQUIREMENTS
9703	USAR-13.06	19	PLANT OPERATIONS - OPERATIONAL REVIEW AND AUDITS
9703	USAR-13.07	19	PLANT OPERATIONS - EMERGENCY PROCEDURES
9703	USAR-13.08	19	PLANT OPERATIONS - REFERENCES
9703	USAR-13.FIG	19	PLANT OPERATIONS - FIGURES
9703	USAR-13.TOC	19	PLANT OPERATIONS - TABLE OF CONTENTS
9703	USAR-14.01	19	PLANT SAFETY ANALYSIS - SUMMARY DESCRIPTION
9703	USAR-14.02	19	PLANT SAFETY ANALYSIS - FUEL CLADDING INTEGRITY SAFETY LIMIT
9703	USAR-14.03	19	PLANT SAFETY ANALYSIS - OPERATING LIMITS
9703	USAR-14.04	19	PLANT SAFETY ANALYSIS - TRANSIENT EVENTS ANALYZED FOR CORE RELOAD
9703	USAR-14.05	19	PLANT SAFETY ANALYSIS - SPECIAL EVENTS

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ______________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-14.06	19	PLANT SAFETY ANALYSIS - PLANT STABILITY ANALYSIS
9703	USAR-14.07	19	PLANT SAFETY ANALYSIS - ACCIDENT EVALUATION METHODOLOGY
9703	USAR-14.08	19	PLANT SAFETY ANALYSIS - ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)
9703	USAR-14.09	19	PLANT SAFETY ANALYSIS - POST-ACCIDENT PLANT SYSTEMS SHIELDING DESIGN REVIEW
9703	USAR-14.10	19	PLANT SAFETY ANALYSIS - OTHER ANALYSES
9703	USAR-14.11	19	PLANT SAFETY ANALYSIS - REFERENCES
9703	USAR-14.FIG	19	PLANT SAFETY ANALYSIS - FIGURES
9703	USAR-14.TOC	19	PLANT SAFETY ANALYSIS - TABLE OF CONTENTS
9703	USAR-14A		NUCLEAR ANALYSIS AND DESIGN
9703	USAR-15.TOC	19	USAR DRAWINGS - TABLE OF CONTENTS
9703	USAR-C	24	OPERATIONAL QUALITY ASSURANCE PLAN
9703	USAR-D.01	19	APPENDIX D PRE-OPERATIONAL AND STARTUP TESTS-TEST PROGRAM SUMMARY DESCRIPTION
9703	USAR-D.02	19	APPENDIX D PRE-OPERATIONAL AND STARTUP TESTS-TEST PROGRAM CONSIDERATIONS
9703	USAR-D.03	19	APPENDIX D PRE-OPERATIONAL AND STARTUP TESTS-CONSTRUCTION TESTS
9703	USAR-D.04	19	APPENDIX D PRE-OPERATIONAL AND STARTUP TESTS-SUMMARY OF PRE-OPERATIONAL TEST CONTENT
9703	USAR-D.05	19	APPENDIX D PRE-OPERATIONAL AND STARTUP TESTS-STARTUP AND POWER TEST PROGRAM
9703	USAR-D.TOC	19	APPENDIX D PRE-OPERATIONAL AND STARTUP TESTS-TABLE OF CONTENTS
9703	USAR-E.01	19	APPENDIX E PLANT COMPARATIVE EVALUATION WITH THE PROPOSED AEC 70 DESIGN CRITERIA-SUMMARY DESCRIP-
9703	USAR-E.02	19	APPENDIX E PLANT COMPARATIVE EVALUATION WITH THE PROPOSED AEC 70 DESIGN CRITERIA-CRITERION-CONFORM-
9703	USAR-E.TOC	19	APPENDIX E PLANT COMPARATIVE EVALUATION WITH THE PROPOSED AEC 70 DESIGN CRITERIA-TABLE OF CONTENTS
9703	USAR-I.01	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-EVALUATION CRITERIA
9703	USAR-I.02	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-HIGH ENERGY SYSTEMS AND PIPING
9703	USAR-I.03	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-BREAK ANALYSIS

Document Distribution Manifest
Controlled Documents - Current

Name: SAM MIRANDA
Title: PROJECT MANAGER NRC
Company: US NRC, MAIL STOP 0-8H2
Location: 1 WHITE FLINT NORTH
11555 ROCKVILLE PIKE
ROCKVILLE, MD 20852-2738

Manifest Number: 2002-426
Distribution Type: I
Holder Number: 67
ID Number: ZZZZ63

Date Mailed: ______________

Distribution Type: UPDATED SAFETY ANALYSIS REPORT(USAR) - MANUAL

NOTE: The documents listed below are new or revised

Document Type	Document Number	Revision	Title
9703	USAR-I.04	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-SAFE SHUTDOWN REQUIREMENTS
9703	USAR-I.05	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-HELB AND SAFE SHUTDOWN EVALUA-
9703	USAR-I.06	19	APPENDIX I EVALAUTION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-REFERENCES
9703	USAR-I.FIG	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-FIGURES
9703	USAR-I.TOC	19	APPENDIX I EVALUATION OF HIGH ENERGY LINE BREAKS OUTSIDE CONTAINMENT-TABLE OF CONTENTS
9703	USAR-LOEP	19	LIST OF EFFECTIVE PAGES
9703	USAR-TOC	19	TABLE OF CONTENTS

MONTICELLO
UPDATED SAFETY ANALYSIS REPORT

Revision 19 Posting Instructions

Location	Remove Pages	Insert Revision 19 Pages
VOLUME 1		
	Transmittal Letter	Transmittal Letter
	Exhibit A	Exhibit A
	Exhibit B	Exhibit B
		Exhibit C
		Exhibit E
Document Control	USAR LOEP	USAR LOEP
Table of Contents	USAR TOC	USAR TOC
Section 1	USAR 1	USAR 1
	USAR 1.1	USAR 1.1
	USAR 1.2	USAR 1.2
	USAR 1.3	USAR 1.3
	USAR 1.4	USAR 1.4
	USAR 1.5	USAR 1.5
	USAR 1.FIGURES	USAR 1.FIGURES
Section 2	USAR 2	USAR 2
	USAR 2.1	USAR 2.1
	USAR 2.2	USAR 2.2
	USAR 2.3	USAR 2.3
	USAR 2.4	USAR 2.4
	USAR 2.5	USAR 2.5
	USAR 2.6	USAR 2.6
	USAR 2.7	USAR 2.7
	USAR 2.8	USAR 2.8
	USAR 2.9	USAR 2.9
	USAR 2.10	USAR 2.10
	USAR.2.FIGURES	USAR.2.FIGURES
Section 3	USAR 3	USAR 3
	USAR 3.1	USAR 3.1
	USAR 3.2	USAR 3.2
	USAR 3.3	USAR 3.3
	USAR 3.4	USAR 3.4
	USAR 3.5	USAR 3.5
	USAR 3.6	USAR 3.6
	USAR 3.7	USAR 3.7
	USAR 3.FIGURES	USAR 3.FIGURES
Section 4	USAR 4	USAR 4
	USAR 4.1	USAR 4.1
	USAR 4.2	USAR 4.2
	USAR 4.3	USAR 4.3
	USAR 4.4	USAR 4.4

Revision 19 Posting Instructions

Location	Remove Pages	Insert Revision 19 Pages
	USAR 4.5	USAR 4.5
	USAR 4.6	USAR 4.6
	USAR 4.7	USAR 4.7
	USAR 4.8	USAR 4.8
	USAR 4.FIGURES	USAR 4.FIGURES
VOLUME 2		
Section 5	USAR 5	USAR 5
	USAR 5.1	USAR 5.1
	USAR 5.2	USAR 5.2
	USAR 5.3	USAR 5.3
	USAR 5.4	USAR 5.4
	USAR 5.FIGURES	USAR 5.FIGURES
Section 6	USAR 6	USAR 6
	USAR 6.1	USAR 6.1
	USAR 6.2	USAR 6.2
	USAR 6.3	USAR 6.3
	USAR 6.4	USAR 6.4
	USAR 6.5	USAR 6.5
	USAR 6.6	USAR 6.6
	USAR 6.7	USAR 6.7
	USAR 6.8	USAR 6.8
	USAR 6.FIGURES	USAR 6.FIGURES
Section 7	USAR 7	USAR 7
	USAR 7.1	USAR 7.1
	USAR 7.2	USAR 7.2
	USAR 7.3	USAR 7.3
	USAR 7.4	USAR 7.4
	USAR 7.5	USAR 7.5
	USAR 7.6	USAR 7.6
	USAR 7.7	USAR 7.7
	USAR 7.8	USAR 7.8
	USAR 7.9	USAR 7.9
	USAR 7.10	USAR 7.10
	USAR 7.11	USAR 7.11
	USAR 7.12	USAR 7.12
	USAR 7.13	USAR 7.13
	USAR 7.14	USAR 7.14
	USAR 7.FIGURES	USAR 7.FIGURES
VOLUME 3		
Section 8	USAR 8	USAR 8
	USAR 8.1	USAR 8.1

Revision 19 Posting Instructions

Location	Remove Pages	Insert Revision 19 Pages
	USAR 8.2	USAR 8.2
	USAR 8.3	USAR 8.3
	USAR 8.4	USAR 8.4
	USAR 8.5	USAR 8.5
	USAR 8.6	USAR 8.6
	USAR 8.7	USAR 8.7
	USAR 8.8	USAR 8.8
	USAR 8.9	USAR 8.9
	USAR 8.10	USAR 8.10
	USAR 8.11	USAR 8.11
	USAR 8.12	USAR 8.12
	USAR 8.13	USAR 8.13
	USAR 8.FIGURES	USAR 8.FIGURES
Section 9	USAR 9	USAR 9
	USAR 9.1	USAR 9.1
	USAR 9.2	USAR 9.2
	USAR 9.3	USAR 9.3
	USAR 9.4	USAR 9.4
	USAR 9.5	USAR 9.5
	USAR 9.FIGURES	USAR 9.FIGURES
Section 10	USAR 10	USAR 10
	USAR 10.1	USAR 10.1
	USAR 10.2	USAR 10.2
	USAR 10.3	USAR 10.3
	USAR 10.4	USAR 10.4
	USAR 10.5	USAR 10.5
	USAR 10.FIGURES	USAR 10.FIGURES
Section 11	USAR 11	USAR 11
	USAR 11.1	USAR 11.1
	USAR 11.2	USAR 11.2
	USAR 11.3	USAR 11.3
	USAR 11.4	USAR 11.4
	USAR 11.5	USAR 11.5
	USAR 11.6	USAR 11.6
	USAR 11.7	USAR 11.7
	USAR 11.8	USAR 11.8
	USAR 11.FIGURES	USAR 11.FIGURES
VOLUME 4		
Section 12	USAR 12	USAR 12
	USAR 12.1	USAR 12.1
	USAR 12.2	USAR 12.2
	USAR 12.3	USAR 12.3

Revision 19 Posting Instructions

Location	Remove Pages	Insert Revision 19 Pages
	USAR 12.4	USAR 12.4
	USAR 12.5	USAR 12.5
	USAR 12.FIGURES	USAR 12.FIGURES
Section 13	USAR 13	USAR 13
	USAR 13.1	USAR 13.1
	USAR 13.2	USAR 13.2
	USAR 13.3	USAR 13.3
	USAR 13.4	USAR 13.4
	USAR 13.5	USAR 13.5
	USAR 13.6	USAR 13.6
	USAR 13.7	USAR 13.7
	USAR 13.8	USAR 13.8
	USAR 13.FIGURES	USAR 13.FIGURES
Section 14	USAR 14	USAR 14
	USAR 14.1	USAR 14.1
	USAR 14.2	USAR 14.2
	USAR 14.3	USAR 14.3
	USAR 14.4	USAR 14.4
	USAR 14.5	USAR 14.5
	USAR 14.6	USAR 14.6
	USAR 14.7	USAR 14.7
	USAR 14.8	USAR 14.8
	USAR 14.9	USAR 14.9
	USAR 14.10	USAR 14.10
	USAR 14.11	USAR 14.11
	USAR 14.FIGURES	USAR 14.FIGURES
14A. Current Cycle's Supplemental Reload Licensing Submittal	"Section 14A Update for Monticello Cycle 20" report	"Section 14A Update for Monticello Cycle 21" report
VOLUME 5		
Section 15	USAR 15 All drawings	USAR 15 All drawings
VOLUME 6		
Appendix A	NO CHANGE	NO CHANGE
VOLUME 7		
Appendix C	Operational Quality Assurance Plan, Rev 22	Operational Quality Assurance Plan, Rev 24

MONTICELLO
UPDATED SAFETY ANALYSIS REPORT

Revision 19 Posting Instructions

Location	Remove Pages	Insert Revision 19 Pages
Appendix D	USAR Appendix D	USAR Appendix D
	USAR D.1	USAR D.1
	USAR D.2	USAR D.2
	USAR D.3	USAR D.3
	USAR D.4	USAR D.4
	USAR D.5	USAR D.5
Appendix E	USAR Appendix E	USAR Appendix E
	USAR E.1	USAR E.1
	USAR E.2	USAR E.2
Appendix F	NO CHANGE	NO CHANGE
VOLUME 8		
Appendix I	USAR APPENDIX I	USAR APPENDIX I
	USAR I.1	USAR I.1
	USAR I.2	USAR I.2
	USAR I.3	USAR I.3
	USAR I.4	USAR I.4
	USAR I.5	USAR I.5
	USAR I.6	USAR I.6
	USAR I.FIGURES	USAR I.FIGURES

If you have questions, please call Jessie at ext. 1230

June 11, 2002

10CFR 50.71(e)
10CFR 50.59(d)(2)

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

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Submittal of Revision No. 19 to the Updated Safety Analysis Report

Pursuant to 10 CFR Part 50, Section 50.71(e), Revision No. 19 to the Updated Safety Analysis Report (USAR) for the Monticello Nuclear Generating Plant is hereby submitted. This revision completes an update of the information in the USAR for the period from June 1, 2000 through March 30, 2002.

A substantial number of the changes reflect the resolution of comments and issues identified as part of the special USAR Review Project, which was initiated in September 1997. The USAR review project was completed in June 2001. The remainder of the changes in this revision reflect the consideration of design changes and safety review items. These changes were made in accordance with the guidance provided in Nuclear Energy Institute (NEI) 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, and R.G. 1.181.

Included as part of this submittal is the periodic report of changes, tests and experiments required by 10 CFR Part 50, Section 50.59(d)(2). The summary report of changes, tests and experiments requiring evaluation under the provisions of 10 CFR 50.59 is provided as Exhibit A.

Exhibit B, "Report of Changes to Licensee Docketed Commitments," provides a brief description and summary of changes to NRC commitments identified to be reported to the Commission in accordance with guidance provided in NEI 99-04, "Guidelines for Managing NRC Commitment Changes." This letter contains no new NRC commitments.

Exhibit C, "Report of Information Removed from the USAR," provides a summary of information removed from the USAR in this revision cycle. This information is provided in accordance with NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1.

Exhibit D contains Revision 19 the Monticello USAR and instructions for posting the document. The USAR is being submitted electronically on CD-Rom according to the instructions in RIS 2001-005, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM."

Exhibit E, "Report of Changes to Monticello Fire Protection Program," provides a summary of changes to the Monticello Fire Protection Program. Changes to the Fire Protection Program are provided in accordance with 10 CFR 50.71(e), 10 CFR 50.59 and the guidance in Generic Letter 86-10.

I hereby certify that I am a duly authorized officer of Nuclear Management Company, LLC, and that to the best of my knowledge, information, and belief, the information provided in the attached Revision 19 to the Monticello USAR meets the requirements of 10 CFR 50.71(e) to update the USAR through March 30, 2002.

Please contact Doug Neve, Licensing Manager, at (763) 295-1353 if you require additional information related to this submittal.

Jeffrey S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

cc: Regional Administrator – III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
J. Silberg (w/o Exhibit C)

Enclosures:	Exhibit A	Monticello Nuclear Generating Plant Report of Changes, Tests and Experiments
	Exhibit B	Monticello Nuclear Generating Plant Report of Changes to Licensee Docketed Commitments
	Exhibit C	Monticello Nuclear Generating Plant Summary of Information Removed from the USAR
	Exhibit D	USAR Revision 19 Changes
	Exhibit E	Report of Changes to Monticello Fire Protection Program

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS

The following sections include a brief description and a summary of the 10CFR50.59 evaluations for those changes, tests and experiments that were carried out without prior NRC approval, pursuant to the requirements of 10 CFR Part 50, Section 50.59(d)(2).

1. <u>Domestic Water Supply to Plant Systems (SRI 00-003)</u>

<u>Description</u>:

The Safety Review Item (SRI) evaluated proposed revisions to Section 10.3.5 of the Updated Safety Analysis Report (USAR) that clarified and provided additional description of the Domestic Water System. Previously, the USAR stated that the Domestic Water System is completely independent of the plant process systems, which was incorrect. It is connected to both the Seal Water System and the Plant Makeup Water Treatment System. This SRI was required because it provided the technical and regulatory justification for the proposed USAR changes.

<u>10CFR50.59 Evaluation Summary</u>:

The description of the Domestic Water System was clarified in USAR Sections 10.3.5.1, 10.3.5.2 and 10.3.5.3. The clarifications state that this system provides raw water to the Plant Makeup Water Treatment System and normal supply for the Seal Water System. These clarifications also provide additional description for the Seal Water supply and a USAR reference for the Plant Makeup Water Treatment System. The revision deleted the incorrect statement that the Domestic Water System is completely independent of the plant process systems. These changes had no effect on the design, function or operation of this system.

2. <u>Automatic Depressurization Systems (ADS) Circuit Description Discrepancies (SRI 98-009)</u>

<u>Description</u>:

This 10CFR50.59 evaluation addressed discrepancies noted in actuation logic and other circuit descriptions between ADS system prints and the USAR description. An NRC resident inspection identified and addressed an incorrect description of the Automatic Depressurization System (ADS) logic in the USAR. Two USAR corrections resulted in the ADS logic circuit being referenced as "*two-out-of-two-once*" rather than "*one-out-of-two twice*." The change was consistent with ADS system prints, Technical Specification Table 3.2.2 and Technical Specification Bases. Two prudent clarifications better described an AC-interlock permissive and an automated power supply switchover scheme. Also, a statement about certain switches being permissive while de-energized was eliminated. This SRI was required to address the non-conforming conditions by performance of a 10CFR50.59 evaluation.

10CFR50.59 Evaluation Summary:

These USAR changes were necessary to make the ADS description in Section 6.2.5.2 reflect the actual system configuration. The statement referencing "permissive when de-energized" switches was removed. The ADS actuation logic descriptions were changed to agree with station prints and Technical Specifications Bases and Table 3.2.2 by changing the "*one-out-of-two-twice*" logic statements to "*two-out-of-two once.*" Two other clarifications better described the power bus transfer scheme and the AC interlock logic associated with the operation of the low pressure pumps. These changes did not represent an unreviewed safety question and provided a more accurate and clearly written USAR.

3. Control Room Ventilation Continuous Use of EFT for Outside Air (SRI 00-016)

Description:

This 10CFR50.59 evaluation addressed the practice of the Control Room Ventilation (CRV) continually drawing outside air through the Emergency Filtration Trains (EFTs) rather than an infrequent use as was described in the USAR. Also, recirculation through the air conditioning units is not the exclusive, normal mode of operation, since fresh outside air is also constantly introduced to the CRV system through the EFTs in conjunction with recirculation.

10CFR50.59 Evaluation Summary:

USAR Section 6.7 changes to incorporate constant use of the EFT for fresh air supply to the control room were acceptable with respect to safety and reliability. The change allowed the CRV system to operate in recirculation mode exclusively or in conjunction with a continuous supply of fresh air. The "normal" mode of operation is described as using both recirculation and supplemental outside air through the EFTs.

4. All Feedwater Pump Trips Not Listed in USAR 7.7.4.2 (SRI 00-004)

Description:

The USAR Review Project identified that not all of the Reactor Feedwater Pump Trips were listed in USAR Section 7.7.4.2. This SRI evaluated the addition of the low suction flow, low lube oil pressure and motor fault pump trips to this USAR section. This SRI was required because it provided the technical and regulatory justification for the proposed USAR change.

10CFR50.59 Evaluation Summary:

The USAR Review Project identified that not all of the Reactor Feedwater Pump Trips were listed in USAR Section 7.7.4.2. SRI 00-004 provided justification for also listing the low suction flow and low lube oil pressure trips, which were part of the pump motors original design, and also for listing the motor fault (overload) trip, which was added by an approved modification. This USAR revision provides a complete listing of the Reactor Feedwater Pump trips.

5. Heavy Loads Definition / Limit (SRI 00-011)

Description:

USAR Section 12.2.5.1 states in part,

A heavy load is defined, by NUREG-0612, as "any load, carried in a given area ..., that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool ...". At Monticello, a heavy load has been conservatively calculated to be any load heavier than 1500 lbs.

Based on the NUREG-0612 definition, the heavy load limit at Monticello was established at 1500 pounds. The combined weight of a single spent fuel assembly and its associated handling tool at Monticello was about 1570 pounds, which included the weight of the handling equipment (e.g., fuel bridge grapple and arm) of approximately 870 pounds and the weight of the fuel bundle of approximately 700 pounds. Based on this value, Monticello chose a more conservative limit of 1500 pounds to use as the heavy load limit. Modification 83MO91, *Refueling Bridge Mast Replacement & Bridge Upgrade,* (Ref. 1) replaced the mast and grapple assembly with a lighter stainless steel assembly. The new assembly weighs approximately 350 pounds. Therefore, the combined weight of a single spent fuel assembly and its associated handling tool is currently about 1050 pounds (i.e., 700 lbs + 350 lbs = 1050 lb).

10CFR50.59 Evaluation Summary:

The purpose of this SRI was to provide a description and evaluation of a USAR change that was identified via the corrective action program. The heavy load limit over the spent fuel pool and the reactor core was conservatively established at 1100 lb and the heavy load limit for the balance of plant remained at 1500 lb. The heavy load limits met the intent of NUREG-0612. They had no impact upon Technical Specifications. They did not involve changes to operation, maintenance or testing. There were no new physical or design changes. The changes met applicable codes and regulations. No new types of hazards, failure modes or interactions were identified. Existing license basis accident analyses and radiation dose calculations remain valid.

6. Torus Cooling During RCIC Operation (SRI 00-005)

Description:

This 10CFR50.59 evaluation addressed the time allowed before torus cooling becomes necessary once the Reactor Core Isolation Cooling (RCIC) turbine exhaust steam begins discharging to the suppression pool. A portion of a statement in USAR Section 10.2.5.2, regarding a four-hour time duration available before torus cooling is necessary with RCIC steam turbine operation, was identified as inaccurate. The text was non-descriptive with respect to the basis for the stated duration and did not clearly indicate whether cooling was to be initiated or the desired effects of cooling occurred at the four-hour mark. The statement was qualitative with dependence on the available heat sink and the corresponding heat production under RCIC operation, but the heat capacity of the suppression pool was variable with its initial temperature, therefore, it was meaningless to declare a specific time available before initiating pool cooling. The existing suppression pool Technical Specifications are temperature based. Temperature is the first order parameter as it governs the effectiveness and capacity of the pool to quench steam. The suppression pool and Residual Heat Removal (RHR) systems are necessary for the safe shutdown of the plant and are safety related. An SRI was necessary to correct this statement.

10CFR50.59 Evaluation Summary:

The USAR Section 10.2.5.2 words, "... after a specified time interval (4 hours)" were removed to eliminate any time dependence and/or allowance to the suppression pool cooling function. This also increased the clarity of the statement by removing the erroneous reference to a time interval that was provided with no supporting basis.

The cooling requirement, as described by Technical Specifications, is governed by temperature, which is the pool parameter most significant (assuming proper volume) with regard to quenching steam. The time to initiate cooling beyond the Technical Specification limit of 90°F is an unfounded limitation, since cooling is provided, per plant procedures, prior to reaching the temperature limit. The referenced emergency procedure is consistent with the NRC approved Emergency Procedure Guidelines.

7. CRD/CRH System Discrepancies in USAR (SRI 00-001)

Description:

The issues addressed by this 10CFR50.59 evaluation dealt primarily with Control Rod Drive/Control Rod Housing (CRD/CRH) system operating parameters and physical configurations that differed from the USAR system description. Many issues from separate USAR sections are directly related to each other since they may be attributed to a particular variation in water pressure or flow, or some other common parameter. The system is operable despite the noted discrepancies and much of the noted variation is a result of incorporating operating experience in designed flow or pressure adjustments. It is, and always has been, a Monticello Nuclear Generating Plant (MNGP) imperative to operate the system in a manner to optimize reliability and performance. All system performance and Technical Specification requirements are met by the CRD/CRH system with its present configuration and operational parameters.

10CFR50.59 Evaluation Summary:

Changes made to CRD/CRH operational parameters were the direct result of adjustments to optimize operation, made possible by operating experience. Other description changes are minor in nature and do not impact the operability of the system.

The changes corrected discrepancies between USAR Section 3.5 text and actual plant parameters and configurations. The CRD/CRH system flows, pressure and associated forces were updated to reflect current operational values. Other descriptions were adjusted to more accurately describe the existing system.

8. Design Basis for RCIC Pump Minimum Flow Bypass Valve Opening Parameter (SRI 00-010)

Description:

A site condition report evaluated a discrepancy with the stroke time for the Reactor Core Isolation Cooling (RCIC) Pump Minimum Flow Bypass Valve as stated in USAR Table 10.2-3. That assessment concluded that the stroke time had been incorrectly listed in the USAR as 15 seconds. This SRI evaluated revising this stroke time to the General Electric design specification value of 5 seconds. This SRI provided the technical and regulatory justification for the proposed USAR change.

10CFR50.59 Evaluation Summary:

Although the allowable stroke time for the RCIC Pump Minimum Flow Bypass Valve listed in USAR Table 10.2-3 was 15 seconds, assessment determined that it should have been listed as 5 seconds. This SRI provided justification for revising the stroke time to 5 seconds. This was a conservative change that did not affect the capability of RCIC to deliver design flow within 30 seconds.

9. Establish Design Temperature Gradient for Spent Fuel Storage Pool Wall (SRI 00-017)

Description:

The purpose of this SRI was to provide a description and evaluation of a USAR change that was identified via the corrective action program. USAR Section 12.2.2.1.1.f, *Thermal Loads*, stated that the design loads used to evaluate the spent fuel pool structure included a temperature gradient of 50°F through slab and walls under normal (operating) conditions. Per USAR Section 10.2.2.1, the spent fuel pool cooling operation at temperatures up to 140°F is acceptable. With the minimum bulk air temperature of 60°F for areas adjacent to the spent fuel pool structure, the temperature gradient through the slab and walls of the spent fuel pool could exceed 50°F.

10CFR50.59 Evaluation Summary:

As a result of this SRI, the spent fuel pool design temperature gradient through slab walls under operating conditions was changed from 50°F to 72°F. This change had no impact on the Technical Specifications. Operations, maintenance and testing were not affected. There were no new physical or design changes. The change met applicable codes and regulations. No new types of hazards, failure modes or interactions were identified. Existing license basis accident analyses and radiation dose calculations remain valid.

10. Incorrect Bypass Delay Time for the RMCS Shutdown Scram (SRI 00-008)

Description:

This 10CFR50.59 evaluation addressed the scram bypass timer being set for two seconds rather than ten seconds as indicated in USAR Section 7.6.1.2.8. The ten-second value dates back to the original FSAR text. The physical setting of the bypass timer in the plant is, and always has been, two seconds.

10CFR50.59 Evaluation Summary:

This SRI was written to justify the change to the Reactor Manual Control System (RMCS) bypass timer description in the USAR from ten to two seconds. There was no physical change performed since the timer is currently set at two seconds. No unreviewed safety question was created by this change.

There was no functional consequence resulting from introduction of the shutdown scram bypass earlier than ten seconds (as was indicated in the USAR). The two-second duration ensures adequate time for the trip relays to de-energize the Reactor Protection System (RPS) seal-in circuit to de-energize the CRD scram pilot valves. The automatic bypass itself is entirely passive with respect to any protective function actuation and its specific delay duration is not relied upon in any accident analysis. The control rods will have been scrammed at such time the RPS is allowed to reset. Since the bypass is a prerequisite for the reset, it is prudent that the bypass timer is set to a shorter duration than the reset timer.

11. Elimination of Design Condition Parameter from USAR Table 5.2-1, "Principal Design Parameters of Primary Containment" (SRI 00-015)

Description:

This SRI evaluated an issue found during the USAR Review Project. The "normal internal pressure" for Primary Containment was listed in USAR Table 5.2-1 as 1.75 psig. No basis for this value was identified. This SRI evaluated revision of the USAR table to show the normal internal pressure range for Primary Containment. The SRI provided the regulatory and technical justification for the proposed USAR change.

10CFR50.59 Evaluation Summary:

It was determined that the "normal internal pressure" for the Primary Containment System, listed in USAR Table 5.2-1 as 1.75 psig, was incorrect and did not have a design basis. This SRI provided justification for deleting this pressure and replacing it with the correct operating pressure range. Operating data showed that the normal internal pressure was less than 1.75 psig and that the Primary Containment high-pressure signal alarm setpoint was also less than this value. This change did not affect the design or accident mitigation capability of the Primary Containment System.

12. Investigate and Resolve USAR Statements Regarding HPCI (SRI 00-002)

Description:

This SRI evaluated an issue found during the USAR Review Project. The 10CFR50.59 evaluation assessed several discrepancies between the USAR and other design documents concerning the High Pressure Coolant Injection (HPCI) turbine design parameters and the HPCI pump discharge isolation valves' position following a turbine trip. The 10CFR50.59 evaluation provided the regulatory and technical justification for proposed USAR changes. These proposed changes affect the HPCI System, specifically the HPCI turbine and both the HPCI pump discharge inboard and outboard isolation valves.

10CFR50.59 Evaluation Summary:

Some of the HPCI turbine design parameters listed in USAR Table 6.2-3 were revised to meet the original values as specified by the HPCI turbine vendor, Terry Turbine. These changes were either insignificant or minor changes, correction of administrative errors, or corrections for consistency between different USAR sections. A USAR statement that the pump discharge valves were prevented from opening automatically whenever a turbine trip condition existed was incorrect and was deleted. There are no design or operational requirements for this. These valves are not required to close to support any safety-related function. The proposed USAR changes do not constitute an unreviewed safety question. Revision of the HPCI turbine design parameters and allowing the HPCI pump discharge valves to open automatically when a HPCI turbine trip exists were acceptable from design, operational and radiological standpoints.

13. Install Blank Flange Downstream of XR-10-4 to Prevent Leakage per Jumper 99-29 (SRI 00-025)

Description:

The Seal Vent to Open Radwaste (ORW) for 12 Recirculation Pump (XR-10-4) leaked by. A temporary modification installed a blank flange on the non-safety related, Open Radwaste side of XR-10-4. An SRI was required because the temporary modification affected a USAR figure. The temporary modification has since been removed.

10CFR50.59 Evaluation Summary:

The Seal Vent to Open Radwaste (ORW) for 12 Recirculation Pump (XR-10-4) leaked by. A blank flange was temporarily installed on the non-safety related Open Radwaste side of XR-10-4. The 50.59 evaluation was required because the temporary modification affected a USAR figure. Since the installation was a temporary modification, the affected USAR drawing was not revised. Addition of the blank flange did not constitute an unreviewed safety question.

14. Reactor Recirculation MG Set and Pump Testing (SRI 00-009)

Description:

During preparation for the reactor pressure vessel hydro a problem occurred with the 11 Reactor Recirculation pump motor generator (11 Recirc MG Set) which caused the unit to trip. To provide for post maintenance testing of the 11 Recirc MG Set, operation of the 12 Recirc MG Set, and recirculation pumps, it was necessary to operate the MG sets and pumps at other than minimum speed. This testing occurred during a refueling outage with the reactor shutdown and the reactor water temperature less than 212°F. A review was conducted to identify the potential conflicts between testing and the USAR or Technical Specifications. In order to run reactor coolant recirculation pumps at higher than minimum speed under existing plant conditions, it was necessary to temporarily bypass the <20% feedwater flow interlock. Bypassing the interlock created a conflict with the Technical Specifications on net positive suction head (NPSH) required for the recirc pump components. This SRI explained how the test was safely accomplished without violating jet pump or reactor recirc pump NPSH limits during circumstances where it was desired to run the recirc pumps above minimum speed.

10CFR50.59 Evaluation Summary:

The proposed activities evaluated in this 10CFR50.59 evaluation, (1) temporarily bypassing the <20% feedwater flow interlock; and (2) operating the Recirc MG sets and pumps at greater than minimum flow with less than 20% feedwater flow available, were acceptable activities and did not constitute an unreviewed safety question. These activities were performed under conditions established in this SRI and implemented under a work order, which controlled the testing of the Recirc MG sets.

15. 10CFR50.59 Evaluation for Differences Between the EOPs and the Design Basis – Defeating the HPCI High Torus Water Level Suction Transfer to Allow Continued HPCI Operation Using the CSTs as a Suction Source (SRI 99-018)

Description:

This 50.59 evaluation addressed revision of Monticello's Emergency Operating Procedures (EOPs) to defeat the HPCI high torus water level suction transfer if torus water temperature exceeds 160°F. The analyses described in the USAR assume transfer of the HPCI suction source to the suppression pool from the Condensate Storage Tank (CST) on high torus water level. This issue was evaluated as a part of a larger effort to evaluate differences between the EOPs and plant design basis.

10CFR50.59 Evaluation Summary:

Defeating the HPCI high torus water level suction transfer if torus water temperature exceeds 160°F did not invalidate the design bases or licensing basis accident analysis assumptions, did not result in consequences more severe than the consequences of taking the USAR actions, did not decrease the effectiveness of the EOPs, and did not constitute an unreviewed safety question. Revising the EOPs to defeat the HPCI high torus water level suction transfer if torus water temperature exceeds 160°F did not constitute an unreviewed safety question and did not adversely affect the ability of the EOPs to mitigate the consequences of any mechanistically credible event.

16. EOP LPCI 5-Minute Seal-in Timer Bypass Switch (Mod 00Q250)

Description:

This modification installed permanent switches to remove the need for booted relay contacts when using the following Emergency Operating Procedure (EOP) support procedures: C.5-3205 (TERMINATE AND PREVENT), C.5-3201 (DEFEAT RCIC PRESSURE AND TEMPERATURE ISOLATIONS), and C.5-3503 (DEFEAT DRYWELL COOLER TRIPS). Evaluation of bypassing the LPCI five-minute bypass timer was accomplished under SRI 01-010. Evaluation of bypassing the RCIC pressure and temperature isolations was accomplished under SRI 01-011. The evaluation of bypassing the ECCS trip of drywell cooling was accomplished under SRI 01-012.

10CFR50.59 Evaluation Summary:

This design change was performed to simplify the procedures for bypassing the EOP support procedures identified above. The bypasses were accomplished using knife switches located on control room panels C-03, C-04, and C-25. The Technical Specifications did require revision as a result of this design change. The EOPs were revised to include the function and usage of the revised bypass circuit. The knife switches did not create an unreviewed safety question as determined by 10CFR50.59 evaluation.

17. MET Wind Sensor Upgrade (Design Change 99Q205)

Description:

The MET wind speed and direction sensors on the primary and backup towers were replaced with new combination sensors.

10CFR50.59 Evaluation Summary:

The replaced wind speed and direction sensors were susceptible to damage by high winds and blowing debris. The new combination sensors are less susceptible to damage and require minimal maintenance and calibration.

18. Convert LS 3063 to Bubbler System (Design Change 00Q135)

Description:

This modification changed the method by which the level switch LS-3063, "Turbine Building Floor Drain Hi Level Alarm", initiating condition is sent to the switch to actuate the high level annunciator. Previously an air capture tube was used. This was converted to an air bubbler system. The air bubbler system was installed under a temporary modification (Jumper/Bypass). This modification allowed the Jumper/Bypass to be removed.

10CFR50.59 Evaluation Summary:

This modification changed the method by which level switch LS-3063 initiating condition is sent to the switch to actuate the high level annunciator. Previously an air capture tube was used. It was converted to an air bubbler system. This modification was non-safety related, non-QA related, non-security related, non-fire related, and did not change the Monticello Technical Specifications. The P&ID for air system and radwaste in Chapter 15 of the USAR required revision. Addition of the new tubing and supports described in the modification did not create an unreviewed safety question.

19. Fuel Zone Level Instrumentation Modification (01Q075)

Description:

The purpose of design change 01Q075 was to improve the reliability of the reactor fuel zone level instruments (LT-2-3-112A and LT-2-3-112B) during accident conditions by changing the instrument reference leg sensing line from the feedwater reference columns to the safeguards columns. In addition to the fuel zone level instruments, several other instruments, such as the reactor low low set pressure transmitters and the ECCS 2/3 core height interlock level switches, were rerouted to the safeguards column. The utilization of the safeguards column for these instruments minimized the potential for these instruments to become inoperable due to the reference leg flashing to steam during accident conditions.

10CFR50.59 Evaluation Summary:

In summary, the purpose of design change 01Q075 was to improve the reliability of the reactor fuel zone level instruments (LT-2-3-112A and LT-2-3-112B) during accident conditions by changing the instrument reference leg sensing line from the feedwater reference columns to the safeguards columns. In addition to the fuel zone level instruments, several other instruments, such as the reactor low low set pressure transmitters and the ECCS 2/3 core height interlock level switches, were rerouted to the safeguards column. The utilization of the safeguards column for these instruments minimized the potential for these instruments to become inoperable due to the reference leg flashing to steam during accident conditions. It was determined that an unreviewed safety question did not exist.

20. Remove Spool Piece from Solid Radwaste System and Install Air Lance Insertion Point (Design Change 00Q325)

Description:

To achieve proper mixing in condensate phase separator tanks T-34A and T-34B, an air lance was inserted through the manhole at the top of the tank to the bottom of the tank and service air was admitted through the air lance. This resulted in a large gap between the manhole cover and the upper rim of the manhole that allowed resin to spray upwards and out of the gap. This modification allowed the insertion of the air lance without having to remove the manhole cover by cutting a hole in the manhole cover and affixing a thick flexible rubber gasket to the hole. A blank flange is installed over the hole when not in use.

10CFR50.59 Evaluation Summary:

Previously the method to achieve proper mixing in condensate phase separator tanks T-34A and T-34B was to insert an air lance through the manhole at the top of the tank to the bottom of the tank and admit plant service air through this air lance. This resulted in a large gap between the manhole cover and the upper rim of the manhole that allowed resin to spray upwards and out of the gap. This modification allowed the insertion of the air lance without having to remove the manhole cover by cutting a hole in the manhole cover and affixing a thick flexible rubber gasket to the hole. The hole is just large enough to pass the air lance through the gasket, thus eliminating the need to remove the manhole cover. This eliminated the large gap between the manhole cover and the upper rim of the manhole and prevented resin and sludge from spraying out of the tank. A blank flange is installed over the hole when not in use. Removal of the spool piece and installing blank flanges on line RWN51-4-HC, as described in the scope of this modification, did not create an unreviewed safety question.

21. Chilled Water Vent Valves and V-CC-10 Bypass (Design Change 98Q170)

Description:

This project modified the steam chase supply cooling coil (V-CC-10) configuration to aid in the monitoring of cooling coil performance and to prevent further performance degradation. This project also added vent valves and piping to the chilled water system to aid system start-up and lay-up evolutions.

10CFR50.59 Evaluation Summary:

To reduce the potential for performance degradation of cooling coil V-CC-10 and the resulting increase in steam chase ambient temperature, the following equipment was installed by this design change: filter upstream of the coil, bypass ductwork and associated isolation damper, dP gage across the coil and filter, temperature indicator, and an inspection platform. Vent piping to aid start up and shutdown of the chilled water system was also added. This modification did not create an unreviewed safety question.

22. Full Steam Dilution Recombiner (Design Change 99Q160)

Description:

Modification 99Q160, Full Steam Dilution Recombiner, improves the operational safety of the recombiner system. The potential for a hydrogen burn or detonation within the system is essentially eliminated by diluting the stoichiometric mixture of hydrogen and oxygen with steam. Steam dilution is accomplished by converting the second stage Steam Jet Air Ejectors (SJAEs) to a non-condensing stage; second stage motive steam remains in the process to dilute the offgas mixture.

10CFR50.59 Evaluation Summary:

Modification 99Q160, Full Steam Dilution Recombiner, improved the operational safety of the recombiner system. The potential for a hydrogen burn or detonation within the system was essentially eliminated by diluting the stoichiometric mixture of hydrogen and oxygen with steam. A 10CFR50.59 evaluation concluded that no unreviewed safety question was created by the modification.

23. RHRSW Motor Cooling Coil Pipe Coupling (Design Change 00Q220)

Description:

The cooling water piping was shortened to allow for the installation of a flex hose between the piping and the cooling water connection at the motor.

10CFR50.59 Evaluation Summary:

The RHR Service Water (RHRSW) cooling coil supply and discharge piping for RHRSW pumps was modified to allow for geometry differences at the cooling coil connections on the spare and inservice pump motors. The cooling water piping was shortened to allow for the installation of a flex hose between the piping and the cooling water connection at the motor. A 10CFR50.59 evaluation concluded that this design change did not present an unreviewed safety question.

24. St. Cloud 115KV Line Designation Change (Mod 00Q330)

Description:

This design change revised site drawings, documents, and the USAR to reflect the 115KV transmission line name change from St. Cloud to Industrial Park.

10CFR50.59 Evaluation Summary:

This design change does not represent an unreviewed safety question. There were no Technical Specification changes. This design change revised site drawings, documents, and the USAR to reflect the 115KV transmission line name change from St. Cloud to Industrial Park.

25. SBGT Makeup Air Improvements (Design Change 00Q270)

Description:

The standby gas treatment (SBGT) system provides, whenever secondary containment isolation conditions exist, a small negative pressure to minimize ground level escape of airborne radioactivity. Filters are provided in the system to remove radioactive particulates, and charcoal adsorbers are provided to remove radioactive halogens. All flow from the SBGT System is released through the offgas vent stack and continuously monitored by the stack gas monitoring system.

The objective of this design change was to minimize filtration bypass. Several items were accomplished to meet this goal, including: (1) removal of the abandoned mixing box, associated ducting, back draft damper, and air supply pressure regulator; (2) blanking of the abandoned outside air makeup duct; (3) addition of additional makeup air capacity from the turbine building.

10CFR50.59 Evaluation Summary:

This design change prevented potential standby gas treatment filtration bypass by increasing the flow of makeup air into the room by removing the mixing air box and associated equipment and increasing the vent area to the room from the turbine building. This design change did not represent an unreviewed safety.

26. Single Loop Recirc Operation with the Idle Loop Discharge Valve Closed (SRI 01-002)

Description:

From time to time it is desired to operate the Monticello Plant with one recirculation loop out of service ("single loop operation"). Previously this was done with the pump discharge valve in the shutdown loop essentially closed. This led to difficulties because the Technical Specifications contain temperature difference limits that preclude pump restart if these limits are exceeded. With the discharge valve essentially closed, loop cooldown resulted in potentially exceeding these limits when the loop is out of service for more than an hour or so. This loop cooldown would, in turn, require the reactor to be shut down and depressurized to a significant extent to allow the idle pump to be restarted, and could result in a reactivity transient. The severity of this transient would depend on the specific conditions at the time.

It is expected that opening the recirculation pump discharge valve on the idle loop and sufficiently increasing the speed of the operating pump will result in sufficient water circulation through the idle loop to prevent significant loop cooldown, thereby allowing restart of the idle loop when desired without having to depressurize the reactor.

The recirculation system does not provide a safety-related function other than that it serves as primary system boundary and as a Low Pressure Coolant Injection (LPCI) injection flow path. The pump discharge valve must close when required by the LPCI system logic.

The purpose of this SRI was to provide justification for fully opening the pump discharge valve in an idle recirculation loop. This required a USAR change because USAR Section 4.3.2.1 makes the statement, "In the event that one pump fails or is shut off, the discharge valve in the inoperative driving loop would be manually closed," with no further qualification or information.

10CFR50.59 Evaluation Summary:

SRI 01-002 has shown that the position of the discharge valve of the idle recirculation pump does not affect the acceptability of single loop operation.

EXHIBIT B

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO LICENSEE DOCKETED COMMITMENTS

The purpose of this exhibit is to provide a brief description and a summary of changes to formally tracked commitments established with the NRC by the Monticello Nuclear Generating Plant. These commitments are being identified and reported to the Commission in accordance with guidance provided in NEI technical report 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes."

1. Monticello Commitment M90001A

Source Document: Monticello Licensee Event Report 89-040, "Failure to Meet Secondary Containment Performance Requirements Due to Design Deficiencies"

Commitment: Place administrative hold on AO-2982 to secure it in closed position when SBGT is required to be operable.

Change: Operating procedures to control AO-2982 in the closed position when SBGT is required to be operable. The commitment does not say a hold tag needs to be placed on the switch, only that an administrative hold must secure AO-2982 in the closed position during SBGT operation. The permanent removal of the hold tags allows Operations to control AO-2982 by procedures and enter LCOs as necessary.

2. Monticello Commitment M84119A

Source Document: Technical Evaluation Report TER-C5506-370– Control of Heavy Loads, January 30, 1984

Commitment: Loads of weight greater than one fuel element (excluding the crane load blocks and associated tackle) shall not be transported directly over spent fuel stored in the spent fuel pool without prior NRC approval.

Change: Loads of weight greater than the weight assumed in the refueling accident analysis for one fuel element and its associated grapple assembly shall not be transported directly over spent fuel stored in the spent fuel pool without prior NRC approval. This commitment is contained in the NRC Safety Evaluation Report which accepted NSP's response to NUREG-0612, Control of Heavy Loads at Nuclear Power Plants – Resolution of Generic Technical Activity A-36, July 1980. The revised commitment meets the intent of the original commitment. The weight limit described in the revised commitment as the weight of one fuel element and its associated grapple assembly is consistent with the heavy load limit over the spent fuel pool established in accordance with NUREG-0612 as discussed in Section 12 of the USAR (Revision 18). The heavy load limit over the spent fuel pool is any load greater than weight of one fuel element and its associated grapple assembly. The revised commitment is also consistent with the refueling accident analysis discussed in Section 14 of the USAR (Revision 18) which is based on the weight of one fuel element and its associated grapple assembly. The refueling accident analysis shows that the activity release due to a drop of a fuel element and its associated grapple assembly onto the reactor core of the spent fuel stored in the spent fuel pool is well below the 10CFR100 limits.

EXHIBIT C

MONTICELLO NUCLEAR GENERATING PLANT
SUMMARY OF INFORMATION REMOVED FROM THE USAR

Consistent with the guidance in Nuclear Energy Institute (NEI) Technical Report 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, and Regulatory Guide 1.181, information removed from the Monticello USAR is summarized below.

- A USAR statement that the High Pressure Coolant Injection (HPCI) pump discharge valves are prevented from opening automatically whenever a turbine trip condition exists was incorrect. These valves are not required to close in support of any safety- related function or operational requirement under HPCI turbine trip conditions. The statement was deleted from Section 6.2.4.2.4
- Section 7.11.2 and Table 7.11-1 are to be deleted because information in these sections was obsolete. The discussion did not reflect the current Loss of Coolant Accident (LOCA) analyses, nor did it reflect the current plant abnormal operating procedures.
- In Section 3.3.3.1, references to total peaking factor were eliminated. Monticello has phased out the use of this term since it is redundant to the use of linear heat generation rate (LHGR).
- Deleted ultrasonic testing (UT) inspection results from Section 3.6.2.1 because the material presented was outdated (from 1994) and did not reflect the current inspection. This information also did not advance the core shroud description of which it was a part.
- The plant was initially constructed with a combined iodine/particulate airborne activity monitor (CIM/CAM sampler). In 1978, this unit was replaced with the present Drywell CAM Particulate Monitor; the iodine channel was removed. References to the CIM were removed from USAR Section 4.3.3.3.
- Deleted references to line D12.5-EF/EB from the High Energy Line Break (HELB) discussion in Appendix I. A plant modification resulted in a pressure reduction in this line, which removed it from the HELB category.
- Deleted the description of how the Main Steam Isolation Valve (MSIV) double disk wedge assembly functions to equalize and minimize seat wear during valve closures from Section 5.2.2.5.3 as a result of a plant alteration. The alteration was installed to address damage to the outboard MSIVs due to "wind-milling" of the valve disks.
- In Section 7.2.2.2, information made obsolete due to complete decoupling of the turbine control and the recirculation flow control systems by a past modification was deleted.
- The portion of Section 13.3.5 that discusses overtime restrictions was eliminated. Overtime restrictions are described by Technical Specification 6.1.F.
- References to the Portable Cement Solidification System referred to in Section 9.4.2.2.2 were removed. This system is not located on-site and is no longer available from the vendor (CNSI). This equipment is not used at Monticello for waste solidification.
- A reference to a vacuum breaker analysis submitted to in response Generic Letter (GL) 88-03, but not used as a basis for the NRC Safety Evaluation Report (SER) on Monticello's response to that letter, was eliminated in Section 5.2.1.2.3. This

information was extraneous because NRC approval of Monticello's response to GL 88-03 was not predicated on this analysis.

- References to responses made by the Dresden facility in response to Atomic Energy Commission (AEC) questions were deleted from Section 7.6.1.2.9.

EXHIBIT E

REPORT OF CHANGES TO MONTICELLO FIRE PROTECTION PROGRAM

This section contains a report of changes to the Monticello Fire Protection Program (FPP) in accordance with the provisions of 10 CFR 50.71(e), 10 CFR 50.59, and Generic Letter (GL) 86-10.

In conformance with GL 86-10, the Updated Fire Hazards Analysis (FHS) and Safe Shutdown Analysis (SSA) are incorporated by reference into the Updated Safety Analysis Report (USAR). These reports underwent significant revision in the course of this USAR revision cycle. Electronic copies of the revised FHA and SSA are provided on the CD-ROM accompanying this submittal.

Amendment 119 to the Monticello Facility Operating License changed license condition C.2.4 to conform to Generic Letter 86-10, and relocated the FPP from the Technical Specifications to a licensee controlled FPP. Electronic copies of the following program documents, as incorporated by reference into the USAR, are provided on the enclosed CD-ROM:

- Fire Protection Program Plan (4 AWI-08.01.00, Revision 0)
- Fire Prevention Practices (4 AWI-08.01.01, Revision 19)

Limiting Conditions for Operation and Surveillance Requirements for Fire Detection and Protection Systems are now defined as Impairments. These were relocated from the Technical Specifications as a part of the aforementioned License Amendment to a site implementing procedure. No changes were made to the Impairments subsequent to their transfer to the implementing procedure.

Consistent with the requirements of the Monticello FPP, a summary of occasions on which more than one fire pump is simultaneously inoperable is to be provided to the NRC with the summary of program changes. On two occasions, three fire pumps were declared inoperable for performance of flow capability testing. During this surveillance, two out of three pumps are considered inoperable when their respective hand-switches are turned to the OFF/STOP position to prevent the pumps from automatically starting when testing the third pump. On both occasions, performance of the flow capability test was completed and the pumps were declared operable in approximately four hours. If the fire pumps had been required during the performance of flow testing, provisions were in place to abort the surveillance and return the pumps to service.

LIST OF EFFECTIVE PAGES

Section	Revision
VOLUME 1	
List of Effective Pages	19
T of C Section	19
1 TOC	19
1.1	19
1.2	19
1.3	19
1.4	19
1.5	19
1.FIGURES	19
2 TOC	19
2.1	19
2.2	19
2.3	19
2.4	19
2.5	19
2.6	19
2.7	19
2.8	19
2.9	19
2.10	19
2.FIGURES	19
3 TOC	19
3.1	19
3.2	19
3.3	19
3.4	19
3.5	19
3.6	19
3.7	19
3.FIGURES	19
4 TOC	19
4.1	19
4.2	19
4.3	19
4.4	19
4.5	19
4.6	19
4.7	19
4.8	19
4.FIGURES	19

LIST OF EFFECTIVE PAGES

Section	Revision
VOLUME 2	
5 TOC	19
5.1	19
5.2	19
5.3	19
5.4	19
5.FIGURES	19
6 TOC	19
6.1	19
6.2	19
6.3	19
6.4	19
6.5	19
6.6	19
6.7	19
6.8	19
6.FIGURES	19
7 TOC	19
7.1	19
7.2	19
7.3	19
7.4	19
7.5	19
7.6	19
7.7	19
7.8	19
7.9	19
7.10	19
7.11	19
7.12	19
7.13	19
7.14	19
7.FIGURES	19
VOLUME 3	
8 TOC	19
8.1	19
8.2	19
8.3	19
8.4	19
8.5	19
8.6	19
8.7	19
8.8	19
8.9	19
8.10	19
8.11	19

LIST OF EFFECTIVE PAGES

Section	Revision
8.12	19
8.13	19
8.FIGURES	19
9 TOC	19
9.1	19
9.2	19
9.3	19
9.4	19
9.5	19
9.FIGURES	19
10 TOC	19
10.1	19
10.2	19
10.3	19
10.4	19
10.5	19
10.FIGURES	19
11 TOC	19
11.1	19
11.2	19
11.3	19
11.4	19
11.5	19
11.6	19
11.7	19
11.8	19
11.FIGURES	19
VOLUME 4	
12 TOC	19
12.1	19
12.2	19
12.3	19
12.4	19
12.5	19
12.FIGURES	19
13 TOC	19
13.1	19
13.2	19
13.3	19
13.4	19
13.5	19
13.6	19
13.7	19
13.8	19
13.FIGURES	19

LIST OF EFFECTIVE PAGES

Section	Revision
14 TOC	19
14.1	19
14.2	19
14.3	19
14.4	19
14.5	19
14.6	19
14.7	19
14.8	19
14.9	19
14.10	19
14.11	19
14.FIGURES	19
APPENDIX 14A	NAD-MN-004, Revision 0

VOLUME 5

15 TOC	19

Drawing Number	Revision
NX-7834-2-2	C
NX-7865-7-1	B
NX-7865-7-2	1
NX-7993-1-1	B
NH-36032 (M-100)	N
NH-36034 (M-103)	AF
NH-36035 (M-104)	AJ
NH-36035-2 (M-104-1)	N
NH-36036 (M-105)	AQ
NH-36037 (M-106)	AP
NH-36038 (M-107)	AA
NH-36038-2 (M-107)	R
NH-36041 (M-110)	BP
NH-36041-2 (M-110-2)	F
NH-36042 (M-111)	AD
NH-36042-2 (M-111-1)	E
NH-36043 (M-137)	AA
NH-36044 (M-138)	AP
NH-36045 (M-139)	W
NH-36046 (M-140)	X
NH-36046-2 (M-140)	D
NH-36047-1 (M-141)	AC
NH-36047-2 (M-141-2)	J
NH-36049-1 (M-131)	AW
NH-36049-2 (M-131)	AD
NH-36049-3 (M-131)	AN
NH-36049-4 (M-131)	T
NH-36049-11 (M-131)	N
NH-36049-12 (M-131)	Q

Drawing Number	Revision
NH-36049-14 (M-131)	AA
NH-36049-15 (M-131)	A
NF-36054 (M-1)	F
NF-36055 (M-2)	Q
NF-36055-1	E
NF-36056 (M-3)	M
NF-36056-1	H
NF-36057 (M-4)	H
NF-36058 (M-5)	C
NF-36059 (M-6)	Q
NF-36060 (M-7)	U
NF-36061 (M-8)	H
NF-36062 (M-9)	C
NF-36063 (M-10)	A
NH-36159 (M-142)	AB
NH-36159-2 (M-142)	E
NF-36175 (E-1)	AE
NF-36222 (C-3)	G
NF-36223 (C-4)	E
NH-36241 (M-115)	AY
NH-36242 (M-116)	AW
NH-36242-1 (M-116-1)	D
NH-36242-2 (M-116-2)	A
NH-36243 (M-117)	AL
NH-36243-1 (M-117)	J
NH-36244 (M-118)	AW
NH-36245 (M-119)	S
NH-36246 (M-120)	BH
NH-36247 (M-121)	BL
NH-36248 (M-122)	AH
NH-36249 (M-123)	AJ
NH-36250 (M-124)	AA
NH-36251 (M-125)	AM
NH-36252 (M-126)	AA
NH-36253 (M-127)	V
NH-36254 (M-128)	AU
NH-36255 (M-129)	AB
NH-36256 (M-135)	AB
NH-36257 (M-136)	S
NH-36258 (M-143)	AY
NH-36263 (M-151)	L
NH-36266 (M-154)	K
NH-36489 (M-800)	AC
NH-36489-2 (M-800-1)	L
NH-36516 (M-810)	Y
NE-36640-2 (E-110-2)	J
NE-36640-3 (E-110-3)	AA
NE-36640-4 (E-110-4)	J
NE-36640-4-2 (E-110-4B)	E
NE-36640-6 (E-110-6)	G
NH-36664 (M-112)	BG

Drawing Number	Revision
NH-36665 (M-811)	CE
NH-36666 (M-812)	AT
NE-36771-4 (E-111-4)	G
NH-36881 (M-145)	S
NH-46162 (M-130)	AV
NH-54817-4	N
NF-54822-2	E
NF-54823-2	G
NF-54823-3	C
NF-54823-4	F
NH-91197	K
NE-93523-3 (E-201-3)	F
NE-93523-4 (E-201-4)	E
NH-94896	N
NH-94897	M
ND-95208	E
ND-95209	S
NH-119062	A
NH-178635	B

VOLUME 6

Page	Revision
APPENDIX A Tab	NA
A-i	4
A-ii	4
A-iii	13
A.1-i	4
A.1-ii	4
A.1-iii	4
A.1-iv	4
A.1-v	4
A.1-1	4
A.1-2	4
A.1-3	4
A.1-4	4
A.1-5	4
A.1-6	4
A.1-7	4
A.1-8	4
A.1-9	4
A.1-10	4
A.1-11	4
A.1-12	4
A.1-13	4
A.1-14	4
A.1-15	4
A.1-16	4
A.1-17	4

Page	Revision
A.1-18	4
A.2-i	4
A.2-ii	4
A.2-iii	4
A.2-iv	4
A.2-1	4
A.2-2	4
A.2-3	4
A.2-4	4
A.2-5	4
A.2-6	4
A.2-7	4
A.2-8	4
A.2-9	4
A.2-10	4
A.2-11	4
A.2-12	4
A.2-13	4
A.2-14	4
A.2-15	13
A.2-16	4
A.2-17	4
A.2-18	18
A.2-19	4
A.2-20	4
A.2-21	4

Page	Revision	Page	Revision
A.2-22	4	A.3-9	4
A.2-23	4	A.3-10	4
A.2-24	4	A.3-11	4
A.2-25	4	A.3-12	4
A.2-26	4	A.3-13	4
A.2-27	4	A.3-14	4
A.2-28	4	A.3-15	4
A.2-29	4	A.3-16	4
A.2-30	4	A.3-17	4
A.2-31	4	A.3-18	4
A.2-32	4	A.3-19	4
A.2-33	4	A.3-20	4
A.2-34	4	A.3-21	4
A.2-35	4	A.3-22	4
A.2-36	4	A.3-23	4
A.2-37	4	A.3.24	4
A.2-38	4	A.4-i	4
A.2-39	4	A.4-ii	4
A.2-40	4	A.4-iii	4
A.2-41	4	A.4-iv	4
A.2-42	4	A.4-1	4
A.2-43	4	A.4-2	4
A.2-44	4	A.4-3	4
A.2-45	4	A.4-4	4
A.2-46	4	A.4-5	4
A.2-47	4	A.4-6	4
A.2-48	4	A.4-7	4
A.2-49	4	A.4-8	4
A.2-50	4	A.4-9	4
A.2-51	4	A.4-10	4
A.2-52	4	A.4-11	4
A.2-53	4	A.4-12	4
A.2-54	4	A.4-13	4
A.2-55	4	A.4-14	4
A.2-56	4	A.4-15	4
A.2-57	4	A.4-16	18
A.2-58	4	A.4-17	18
A.2-59	4	A.4-18	18
A.2-60	4	A.4-19	4
A.3-i	4	A.4-20	4
A.3-ii	4	A.4-21	4
A.3-ii	4	A.4-22	4
A.3-iv	4	A.4-23	18
A.3-1	4	A.4-24	4
A.3-2	4	A.4-25	4
A.3-3	4	A.4-26	4
A.3-4	4	A.4-27	4
A.3-5	4	A.4-28	4
A.3-6	4	A.4-29	4
A.3-7	4	A.4-30	4
A.3-8	4	A.4-31	4

Page	Revision
A.4-32	4
A.4-33	4
A.4-34	4
A.4-35	4
A.4-36	4
A.4-37	4
A.4-38	4
A.4-39	4
A.4-40	4
A.4-41	4
A.4-42	4
A.4-43	4
A.4-44	4
A.4-45	4
A.4-46	4
A.4-47	4
A.4-48	4
A.5-i	4
A.5-ii	4
A.5-iii	4
A.5-1	4
A.5-2	4
A.5-3	4
A.5-4	4
A.5-5	4
A.5-6	4
A.5-7	4
A.5-8	4
A.5-9	4
A.5-10	4
A.5-11	4
A.5-12	4
A.5-13	4
A.5-14	4
A.5-15	4
A.5-16	4
A.5-17	4
A.5-18	4
A.5-19	4
A.5-20	4
A.5-21	4
A.5-22	4
A.5-23	4
A.5-24	4
A.5-25	4
A.5-26	4
A.5-27	4
A.5-28	4
A.5-29	4
A.5-30	4
A.5-31	4

Page	Revision
A.5-32	4
A.5-33	4
A.5-34	4
A.5-35	4
A.6-i	4
A.6-1	4
A.6-2	4
A.6-3	4
A.6-4	4
A.6-5	4
A.6-6	4
A.6-7	4
A.6-8	4
A.6-9	4
A.6-10	4
A.6-11	4
A.6-12	4
A.6-13	4
A.6-14	4
A.6-15	4
A.6-16	4
A.6-17	4
A.6-18	4
A.6-19	4
A.6-20	4
A.6-21	4
A.6-22	4
A.6-23	4
A.7-i	4
A.7-1	4
A.7-2	4
A.7-3	4
A.7-4	4
A.7-5	4
A.7-6	4
A.7-7	4
A.7-8	4
A.7-9	4
A.7-10	4
A.7-11	4
A.8-i	4
A.8-1	4
A.8-2	4
A.8-3	4
A.8-4	4
A.8-5	4
A.8-6	4
A.9-i	4
A.9-ii	4
A.9-iii	13
A.9-1	4

Page	Revision
A.9-2	4
A.9-3	4
A.9-4	13
A.9-5	4
A.9-6	4
A.9-7	4
A.9-8	4
A.9-9	4
A.9-10	4

- VOLUME 7 -

Page	Revision
APPENDIX C Tab	NA
Operational QA Plan Rev 24	
Cover Sheet	
Table of Contents	
Pages 1 through 102	
APPENDIX D Tab	NA
D TOC	19
D.1	19
D.2	19
D.3	19
D.4	19
D.5	19
APPENDIX E Tab	NA
E TOC	19
E.1	19
E.2	19
APPENDIX F Tab	NA
F-i	18
F-ii	4
F.1-1	4
F.1-2	4
F.2-1	4
F.2-2	4
F.2-3	4
F.2-4	4
F.2-5	4
F.2-6	4
F.2-7	4
F.2-8	18
FIG F.2.1	4
FIG F.2.2	4
F.3-1	4

Page	Revision
F.4-1	4
F.4.2	4
F.A-i	4
F.A-1	4
F.A-2	4
F.A-3	4
F.A-4	4
F.A-5	4
F.A-6	4
F.A-8	4
F.A-9	4
F.A-10	4
F.A-11	4
F.A-12	4
F.A-13	4
F.A-14	4
F.A.15	4
F.A.A-i	4
F.A.A-1	4
F.A.B-i	4
F.A.B-1	4
F.A.B-2	4
F.A.B-3	4
F.A.B-4	4
F.A.B-5	4
F.A.C-i	4
F.A.C-1	4
F.A.D-i	4
F.A.D-1	4
F.A.D-2	4
F.A.D-3	4
F.A.E-i	4
F.A.E-1	4
F.A.E-2	4
F.A.E-3	4
F.A.E-4	4
F.A.E-5	4
F.A.E-6	4
F.A.F-i	4
F.A.F-1	4
F.A.F-2	4
F.A.F-3	4
F.A.F-4	4
F.A.F-5	4
F.A.F-6	4
F.A.F-7	4
F.A.F-8	4
F.A.F-9	4
F.A.F-10	4
F.A.F-11	4
F.B-i	4

Page	Revision
F.B-1	4
F.B-2	4
F.B-3	4
F.B-4	4
F.B-5	4
F.B-6	4

VOLUME 8

Page	Revision
APPENDIX G Tab	NA
G-i	4
G-ii	4
G-iii	4
G-iv	4
G.1-1	4
G.1-2	4
G.1-3	4
G.2-1	4
G.2-2	4
G.2-3	11
G.2-4	4
G.2-5	4
G.2-6	13
G.3-1	4
G.3-2	4
G.3-3	4
G.3-4	4
G.3-5	4
G.3-6	4
G.3-7	4
G.3-8	4
G.3-9	4
G.4-1	4
G.4-2	4
G.4-3	4
G.4-4	13
G.4-5	4
G.5-1	4
EX 1	4
EX 2	4
EX 3	4
EX 4	4
EX 5	4
EX 6	4
EX 7	4
EX 8	4
EX 9	4

Page	Revision
APPENDIX H Tab	NA
H-i	4
H-ii	5
H.1-1	4
H.1-2	7
H.1-3	7
H.1-4 (blank)	7
Exhibit 1	
1-1	4
Specification 21A112, Rev. 6 pages 1-2 through 1-36	
Attachment A to Specification 21A112, Rev. 3, pages 1-37	
Attachment B to Specification 21A112, Rev. 3, pages 1-38 through 1-43	
Attachment C to Specification 21A112, Rev. 0, pages 1-44 through 1-49	
Attachment D to Specification 21A112, Rev. 1, pages 1-50 through 1-52	
Attachment E to Specification 21A112, Rev. 1 pages 1-53	
Exhibit 2	
2-1	4
2-2	4
2-3	4
2-4	4
2-5a	4
2-5b	4
2-6	4
2-7	4
2-8	4
2-9	4
2-10	4
2-11	4
2-12	4
2-13	4
2-14	4
2-15	4
2-16	4
2-17	4
2-18	4
2-19	4
2-20	4

Page	Revision
2-21	4
2-22	4
2-23	4
2-24	4
2-25	4
2-26	4
2-27	4
2-28	4
2-29	4
2-30	4
2-31	4
2-32	4
Exhibit 3	
3-1	4
3-2	4
3-3	4
3-4	4
3-5	4
3-6	4
3-7	4
3-8	4
3-9	4
3-10	4
3-11	4
3-12	4
3-13	4
3-14	4
3-15	4
3-16	4
3-17	4
3-18	4
3-19	4
3-20	4
3-21	4
3-22	4
3-23	4
3-24	4
3-25	4
3-26	4
3-27	4
3-28	4
3-29	4
3-30	4
3-31	4
3-32	4
Exhibit 4	
4-1	4
4-2	4
4-3	4
4-4	4

Page	Revision
4-5	4
4-6	4
4-7	4
4-8	4
4-9	4
4-10	4
4-11	4
4-12	4
4-13	4
4-14	4
4-15	4
4-16	4
4-17	4
4-18	4
4-19	4
4-20	4
4-21	4
4-22	4
4-23	4
4-24	4
4-25	4
4-26	4
4-27	4
4-28	4
4-29	4
4-30	4
4-31	4
4-32	4
4-33	4
4-34	4
4-35	4
4-36	4
Exhibit 5	
5-1	4
5-2	4
5-3	4
5-4	4
5-5	4
5-6	4
5-7	4
5-8	4
5-9	4
5-10	4
5-11	4
5-12	4
5-13	4
5-14	4
5-15	4
5-16	4
5-17	4

Exhibit 6	
6-1	4
6-2	4
6-3	4
Exhibit 7	
7-1	5
Specification 22A5541, Rev. 2, pages 7-2 through 7-19	
Exhibit 8	
8-1	5
Specification 22A7454, Rev. 1, pages 8-2 through 8-189	
Exhibit 9	
9-1	5
Specification 22A7227, Rev. 0, pages 9-2 through 9-18	
APPENDIX I Tab	NA
I TOC	19
I.1	19
I.2	19
I.3	19
I.4	19
I.5	19
I.6	19
I.FIGURES	19

Revision (Rev) No.	Effective Date	Date of Issue	Remarks
ORIGINAL	10/81	5/1/82	
1	10/82	4/29/83	
2	10/83	4/30/84	
3	12/84	6/28/85	
4	12/85	6/30/86	Added Appendices A thru I
5	12/86	6/30/87	
6	12/87	6/30/88	
7	12/88	6/30/89	
8	11/89	11/4/89	Modified Containment Isolation Valve List to Coordinate Removal of the List from the Tech Spec
9	12/89	6/29/90	
10	12/90	6/28/91	
11	12/91	6/30/92	
12	9/93	9/1/93	Reprint of USAR sections 1 through 14 due to placement of these sections on the Monticello Site Technical Publishing System
13	5/95	4/20/95	Reprint of USAR Appendix D, E, and I due to placement of these sections on the Monticello Site Technical Publishing System
14	5/96	11/18/96	
15	7/97	10/31/97	Revision to Section 5.2 response to Treatment of Commitments as License Conditions (TAC No. 97781) and Tech Spec Amendment 98 license condition.
16	7/98	10/23/98	General revision to reflect periodic update and initial input from USAR Review Project.
17	8/99	8/25/99	Revise sections affected by the implementation of the power rerate approved by License Amendment 102.
18	5/00	8/25/00	General revision to reflect periodic update and continuing input from USAR Review Project.

Revision (Rev) No.	Effective Date	Date of Issue	Remarks
19	6/02		General revision to reflect periodic update and continuing input from USAR Review Project.

Prepared by M Miszczak

Approved by Site Vice President or Designee

FOR
Approved by D Neve

I/jmr

TABLE OF CONTENTS

VOLUME 1

SECTION 1 INTRODUCTION AND SUMMARY

1.1 Purpose, Scope and Organization of Report

1.2 Principal Design Criteria

1.3 Summary Design Description and Safety Analysis

1.4 Identification and Qualification of Contractors

1.5 References

1.FIGURES

SECTION 2 SITE AND ENVIRONS

2.1 Introduction

2.2 Site Description

2.3 Meteorology

2.4 Hydrology

2.5 Geology and Soil Investigation

2.6 Seismology

2.7 Radiation Environmental Monitoring Program (REMP)

2.8 Ecological and Biological Studies

2.9 Consequences of Hypothetical Local Catastrophes

2.10 References

2.FIGURES

SECTION 3 REACTOR

3.1 General Summary

3.2 Thermal and Hydraulic Characteristics

TABLE OF CONTENTS (CONT'D)

3.3 Nuclear Characteristics

3.4 Fuel Mechanical Characteristics

3.5 Reactivity Control Mechanical Characteristics

3.6 Other Reactor Vessel Internals

3.7 References

3.FIGURES

SECTION 4 REACTOR COOLANT SYSTEM

4.1 Summary Description

4.2 Reactor Vessel

4.3 Recirculation System

4.4 Reactor Pressure Relief System

4.5 Reactor Coolant System Vents

4.6 Hydrogen Water Chemistry

4.7 Zinc Water Chemistry (GEZIP)

4.8 References

4.FIGURES

VOLUME 2

SECTION 5 CONTAINMENT SYSTEM

5.1 Summary Description

5.2 Primary Containment System

5.3 Secondary Containment System and Reactor Building

5.4 References

5.FIGURES

TABLE OF CONTENTS (CONT'D)

SECTION 6 PLANT ENGINEERED SAFEGUARDS

6.1 Summary Description

6.2 Emergency Core Cooling System (ECCS)

6.3 Main Steam Line Flow Restrictions

6.4 Control Rod Velocity Limiters

6.5 Control Rod Drive Housing Supports

6.6 Standby Liquid Control System

6.7 Main Control Room, Emergency Filtration Train Building and Technical Support Center Habitability

6.8 References

6.FIGURES

SECTION 7 PLANT INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

7.2 Reactor Control Systems

7.3 Nuclear Instrumentation System

7.4 Reactor Vessel Instrumentation

7.5 Plant Radiation Monitoring Systems

7.6 Plant Protection System

7.7 Turbine-Generator System Instrumentation and Control

7.8 NUMAC Rod Worth Minimizer and Plant Process Computer

7.9 Other Systems Control and Instrumentation

7.10 Seismic and Transient Performance Instrumentation Systems

7.11 Reactor Shutdown Capability

7.12 Detailed Control Room Design Review

TABLE OF CONTENTS (CONT'D)

7.13 Safety Parameter Display System

7.14 References

7.FIGURES

VOLUME 3

SECTION 8 PLANT ELECTRICAL SYSTEMS

8.1 Summary

8.2 Transmission System

8.3 Auxiliary Power System

8.4 Plant Standby Diesel Generator Systems

8.5 DC Power Supply Systems

8.6 Reactor Protection System Power Supplies

8.7 Instrumentation and Control AC Power Supply Systems

8.8 Electrical Design Considerations

8.9 Environmental Qualification of Safety-Related Electrical Equipment

8.10 Adequacy of Station Electrical Distribution System Voltages

8.11 Power Operated Valves

8.12 Station Blackout

8.13 References

8.FIGURES

SECTION 9 PLANT RADIOACTIVE WASTE CONTROL SYSTEMS

9.1 Summary Description

9.2 Liquid Radwaste System

9.3 Gaseous Radwaste System

TABLE OF CONTENTS (CONT'D)

	9.4	Solid Radwaste System
	9.5	References
	9.FIGURES	
SECTION 10	PLANT AUXILIARY SYSTEMS	
	10.1	Summary Description
	10.2	Reactor Auxiliary Systems
	10.3	Plant Service Systems
	10.4	Plant Cooling System
	10.5	References
	10.FIGURES	
SECTION 11	PLANT POWER CONVERSION SYSTEMS	
	11.1	Summary Description
	11.2	Turbine-Generator System
	11.3	Main Condenser System
	11.4	Main Turbine Bypass System
	11.5	Circulating Water System
	11.6	Cooling Tower System
	11.7	Condensate Demineralizer System
	11.8	Condensate and Reactor Feedwater Systems
	11.FIGURES	
VOLUME 4		
SECTION 12	PLANT STRUCTURES AND SHIELDING	
	12.1	Summary Description
	12.2	Plant Principal Structures and Foundations

TABLE OF CONTENTS (CONT'D)

12.3 Shielding and Radiation Protection

12.4 Radioactive Materials Safety

12.5 References

12.FIGURES

SECTION 13 PLANT OPERATIONS

13.1 Summary Description

13.2 Organization, Responsibilities, and Qualifications

13.3 Personnel Experience and Training

13.4 Operational Procedures

13.5 Operational Records and Reporting Requirements

13.6 Operational Review and Audits

13.7 Emergency Procedures

13.8 References

13.FIGURES

SECTION 14 PLANT SAFETY ANALYSIS

14.1 Summary Description

14.2 Fuel Cladding Integrity Safety Limit

14.3 Operating Limits

14.4 Transient Events Analyzed for Core Reload

14.5 Special Events

14.6 Plant Stability Analysis

14.7 Accident Evaluation Methodology

14.8 Anticipated Transients Without Scram (ATWS)

14.9 Post-Accident Plant Systems Shielding Design Review

TABLE OF CONTENTS (CONT'D)

14.10 Other Analyses

14.11 References

14.FIGURES

14.A Current Cycle's Supplemental Reload Licensing Submittal

VOLUME 5

SECTION 15 USAR DRAWINGS

VOLUME 6

APPENDIX A Seismic Design Criteria

APPENDIX B Deleted

VOLUME 7

APPENDIX C Operational Quality Assurance Plan

APPENDIX D Pre-Operational and Startup Tests

APPENDIX E Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria

APPENDIX F Containment Vessel Design Summary Report

VOLUME 8

APPENDIX G Probable Maximum Flood

APPENDIX H Reactor Pressure Vessel Design Summary Report

APPENDIX I Evaluation of High Energy Line Breaks Outside Containment

TABLE OF CONTENTS

Section		Page
1.1	**Purpose, Scope and Organization of Report**	**1**
1.1.1	Introduction	1
1.1.2	Methods of Technical Presentation	2
1.1.2.1	Purpose	2
1.1.2.2	Radioactive Material Barrier Concept	2
1.1.2.3	Organization of Contents	3
1.1.2.4	Format Organization of Subsections	4
1.2	**Principal Design Criteria**	**1**
1.2.1	General Criteria	1
1.2.2	Reactor Core	1
1.2.3	Reactor Core Cooling	2
1.2.4	Plant Containment	2
1.2.5	Plant Instrumentation and Control	3
1.2.6	Plant Electrical Power	3
1.2.7	Plant Radioactive Waste Disposal	3
1.2.8	Plant Shielding and Access Control	4
1.2.9	Plant Fuel Handling and Storage	4
1.2.10	Separation of Safety Systems	4
1.2.11	Class I Equipment and Structures	4
1.2.12	Class II Equipment and Structures	4
1.3	**Summary Design Description and Safety Analysis**	**1**
1.3.1	Plant Site and Environs	1
1.3.1.1	Description of the Site	1
1.3.1.2	Description of the Environs	1
1.3.1.3	Geology	2
1.3.1.4	Hydrology	2
1.3.1.5	Regional and Site Meteorology	2
1.3.1.6	Seismology and Design Response Spectra	3
1.3.1.7	Environmental Monitoring Program	3
1.3.2	Reactor System	3

1.3.3	Plant Containment System	5
1.3.3.1	Primary Containment System	5
1.3.3.2	Secondary Containment System	6
Section		Page
1.3.4	Plant Auxiliary and Standby Cooling Systems	7
1.3.5	Plant Instrumentation Control Systems	8
1.3.5.1	Reactor Control	8
1.3.5.2	Reactor Protection System	8
1.3.5.3	Plant Radiation Monitoring System	9
1.3.6	Plant Fuel Storage and Handling Systems	9
1.3.7	Plant Main Turbine System	10
1.3.8	Plant Electrical Power Systems	10
1.3.9	Plant Shielding, Access Control, and Radiation Protection Procedures	10
1.3.10	Plant Radioactive Waste Control Systems	11
1.3.11	Summary Evaluation of Plant Safety	12
1.3.11.1	General	12
1.3.11.2	Summary of Offsite Doses	12
1.4	**Identification and Qualification of Contractors**	**1**
1.4.1	Licensee	1
1.4.2	Contractors	1
1.5	**References**	**1**
1.FIGURES		**1**
1.3-1	Nuclear Steam Supply Heat Balance 1775 MWt (631.2 MWe)	2
1.3-2	Turbine Generator System - Heat Balance 1775 MWt (631.2 MWe)	3
1.3-3	Plant Process Flow Diagram	4

SECTION 1 INTRODUCTION AND SUMMARY

1.1 Purpose, Scope and Organization of Report

1.1.1 Introduction

This Updated Safety Analysis Report (USAR) is submitted by Nuclear Management Company, LLC, herein designated as NMC, for the Monticello Nuclear Generating Plant, herein designated as the plant, in accordance with the requirements of 10CFR50 Section 50.71(e) as published in the Federal Register on May 9, 1980. 01-005

The Monticello Nuclear Generating Plant (MNGP) is owned by Northern States Power Company (NSP). NSP is a wholly owned utility operating subsidiary of Xcel Energy Corporation (Xcel Energy) (Reference 3). Transfer of operating authority for the plant from NSP to NMC was approved by the Nuclear Regulatory Commission (NRC) in License Amendment 110 (Reference 4). 01-010 01-005

This USAR is the updated version of the Final Safety Analysis Report (FSAR). The FSAR was originally submitted on November 8, 1968. That document will be referred to as the "original FSAR". The last amendment to the FSAR (number 28) was submitted on July 23, 1970. Following July 23, 1970 the FSAR was not amended and became a historical document. This document will be referred to as the "FSAR". The USAR contains a current description of the Monticello Nuclear Generating Plant as of the latest revision date (see "Document Control" Section). This document will be revised per 10CFR50 Section 50.71(e).

The Monticello Nuclear Generating Plant, Unit 1, uses a single cycle, forced circulation, low power density boiling water reactor. General Electric Company designed the plant and supplied the nuclear steam supply system, the initial reactor fuel, and turbine-generator unit and its related systems. This design is identified as "BWR-3" by General Electric. Bechtel Corporation constructed the plant. 01-005 01-010

The plant was constructed, pursuant to Construction Permit CPPR-31, at the Monticello site in Wright County, Minnesota.

Construction started on June 19, 1967, and initial fuel loading was completed during the fall of 1970. Following a period of testing, full commercial operation began on June 30, 1971 under Provisional Operating License Number DPR-22. The Full Term Operating License was issued on January 9, 1981. The Monticello license expires midnight September 8, 2010 per Amendment No. 53.

This USAR contains an analysis and evaluation of the plant, including the core, based on operation at 1670 MWt, which is equivalent to a nominal gross electrical output of 575.4 MWe. The Facility Description and Safety Analysis Report submitted in support of the application for Construction Permit CPPR-31 evaluated the design of the major systems and components of the plant, including the containment and engineered safeguards, at a power level of 1674 MWt. The thermal and hydraulic characteristics of the core, however, were evaluated at a power level of 1469 MWt, the then contemplated initial power level for which an operating license would be sought. Based on more recently developed critical heat flux correlations, the final fuel design, and the radioactivity release rates related to operation at 1670 MWt, authorization was requested for operation of power levels up to 1670 MWt. The startup and test program set forth in Appendix D provided a stepwise increase in power levels up to 1670 MWt and the criteria which were met prior to proceeding to operation at each of the succeeding step increases in power level.

In 1998, the thermal power level was increased to 1775 MWt. Implementation of this rerate power level involved a power ascension test program which took into account applicable elements of the startup test program set forth in Appendix D. A summary report was submitted to the NRC on February 18, 1999 (Reference 2).

1.1.2 Methods of Technical Presentation

1.1.2.1 Purpose

This USAR contains the changes necessary to reflect significant information and analyses submitted to the Commission or prepared by NMC or its predecessor, NSP, pursuant to Commission requirements since the submission of the original FSAR. 01-005

1.1.2.2 Radioactive Material Barrier Concept

Because the safety aspects of this report pertain to the relationship between plant behavior under a variety of circumstances and the radiological effects on persons off-site, the report is oriented to the radioactive material barrier approach. This orientation facilitates evaluation of the radiological effects of the plant on the environment and to the health and safety of the general public.

The overriding consideration that determines the depth of detailed technical information presented about a particular system or component is the relationship of the system or component to the radioactive material barriers. Systems that must operate to preserve the radioactive material barriers are described in the greatest detail. Systems that have little relationship to the radioactive material barriers are described only with as much detail as is necessary to establish their functional role in the plant.

1.1.2.3 Organization of Contents

The USAR Master Table of Contents is provided with Volume One and shows 2-digit subsection detail only. Tables of Contents for each section are provided at the beginning of each section.

The USAR is organized into 15 major sections each of which consists of a number of subsections. Tables are included within and at the end of the subsection in which they are referred to. References are listed in a separate subsection, just ahead of the figures at the end of each major section. Some of the references used in the USAR may not have been previously submitted to the NRC, but most are available from NMC. There are some instances where 01-005 references were included in the USAR because they were listed in the FSAR, but a copy of the reference was not originally obtained from the vendor or are no longer available. These references are generally related to historical information and are not relied on as a basis for demonstrating the design adequacy of the MNGP.

The listing of effective pages showing the current revision of each subsection is provided with Volume One. To the maximum extent practicable, the current revision of each subsection of the appendices is also shown in Volume One; however, in some cases an appendix may be revised in its entirety at each revision and subsection revision control is not applicable.

The principal architectural and engineering criteria which define the broad frame of reference within which the plant is designed are set forth in subsection 1.2. Subsection 1.3 presents a brief description of the site environs and key plant systems.

Sections 2 through 13 present detailed information about the design and operation of the plant. In these sections nuclear safety systems and engineered safeguards are integrated into sections according to system function (emergency core cooling, control, etc.), system type (electrical, mechanical, etc.), or according to their relationship to a particular radioactive material barrier (primary containment, secondary containment, etc.). Section 3, "Reactor", describes plant components and presents design details that are most pertinent to the fuel barrier. Section 4, "Reactor Coolant System," describes plant components and systems that are most pertinent to the reactor system process barrier. Section 5, "Containment Systems", describes the primary and secondary containment systems. Thus, Sections 3, 4, and 5 represent the first four of seven plant radioactive material barriers.

The remainder of the sections group "system" information according to plant function (radioactive waste control, emergency core cooling, power conversion, control, etc.) or system type (electrical, structures, etc.). Subsections presenting information on topics other than plant systems or components are arranged individually according to the subject matter so that the relationship between the subject and public safety is emphasized.

Section 14, "Plant Safety Analysis," provides an overall safety evaluation of the plant which demonstrates both the adequacy of equipment designed to protect the radioactive material barriers and the ability of the safeguard features to minimize or mitigate the consequences of situations in which one or more radioactive material barriers are assumed damaged for analytical purposes.

Section 15, "USAR Drawings" provides a consolidation of drawings which are referred to in various subsections of the USAR.

The appendices to the FSAR describe and evaluate (a) the site and structures' seismic design criteria, (b) the site meteorology and limits, (c) quality assurance programs, (d) plant start-up program, (e) conformance to the AEC's proposed 70 design criteria, (f) the containment vessel design report, and (g) AEC questions and responses. A separate report describes fabrication, erection and testing of the reactor vessel. Each of these documents has been included in the USAR either by reference or actual text incorporation, as appropriate.

Incorporated into the design of this plant are features to improve both operational performance and overall safety which have been presented in special topical reports.

1.1.2.4 Format Organization of Subsections

Subsections are numerically identified by representing their order of appearance in a section by two numbers separated by a decimal point, e.g., 3.4, the fourth subsection in Section 3. Subsections are further subdivided by numbers separated by decimal points (3.4.1.1, etc.). Pages within each subsection are consecutively numbered.

Tabulations of data are designated "Tables" and are identified by the subsection number followed by a hyphen and the number of the table, e.g., Table 3.4-5. Pictures, sketches, curves, graphs, and engineering diagrams are identified as "Figures" and are numbered in the same manner as tables. Drawings are referred to by NSP drawing number and are contained in Section 15.

The general organization of a subsection describing a system or component usually follows:

a. Design Basis

b. Description

c. Performance Analysis

d. Inspection and Testing

1.2 Principal Design Criteria

The principal criteria for design, construction, and operation of this plant are summarized below.[1]

1.2.1 General Criteria

a. The plant is designed, fabricated, erected, and operated to produce electrical power in a safe, reliable and efficient manner and in accordance with applicable codes and regulations.

b. The plant is designed in such a way that the release of radioactive materials to the environment is limited, so that the limits and guideline values of published regulations pertaining to the release of radioactive materials are not exceeded.

c. The design of those components which are important to the safety of the plant includes allowances for the appropriate environmental phenomena at the site. Those components important to safety and required to operate during accident conditions are designed to operate in the post accident environment.

1.2.2 Reactor Core

a. The reactor core is designed as a boiling water reactor to produce steam for direct use in a turbine-generator.

b. The reactor core, in conjunction with other design parameters, is designed so there is no inherent tendency for sudden divergent oscillation of operating characteristics in any mode of operation.

c. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

d. Power excursions which could result from any credible reactivity addition accident do not cause damage, either by motion or rupture, to the reactor vessel or impair operation of required safeguards.

e. The reactor core is designed so that control rod action, with the maximum worth control rod fully withdrawn and unavailable for use, is capable of bringing the reactor core subcritical and maintaining it so from any power level in the operating cycle.

FOR ADMINISTRATIVE USE ONLY			
Resp Supv: TOPE	Assoc Ref: USAR-MAN	SR: N	Freq: 2 yrs
ARMS: USAR-01.02	Doc Type: 9703	Admin Initials: LF	Date: 01/13/09

1. For a comparative evaluation of the Monticello plant with the AEC's proposed 70 General Design Criteria, refer to Appendix E of the USAR.

f. Redundant backup reactivity shutdown capability is provided independent of normal reactivity provisions. This system has the capability, with adequate margin, to shut down the reactor from any power level in the operating cycle.

g. The fuel rod cladding is designed to contain the fission gas released from the fuel material throughout the design life of the fuel rod.

h. Thermal characteristics of the reactor core are adequate to prevent fuel clad surface heat flux or fuel material center temperatures which could cause sudden fuel cladding ruptures.

i. The reactor core and associated systems are designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage.

1.2.3 Reactor Core Cooling

a. Heat removal systems are provided to remove heat generated in the reactor core for the full range of normal operational conditions from plant shutdown to maximum thermal output. The capacity of such systems is adequate to prevent fuel clad damage.

b. Heat removal systems are provided to remove decay heat generated in the reactor core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel clad damage.

c. Redundant heat removal systems are provided to preserve reactor core heat transfer geometry following various postulated design basis loss-of-coolant accidents.

d. Independent means are provided to prevent overpressure conditions which could jeopardize the integrity of the reactor primary system or reactor core cooling systems.

1.2.4 Plant Containment

a. The primary containment system is designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture, or equivalent failure of any coolant pipe within the primary containment.

b. Provision is made both for the removal of energy from within the primary containment and/or such other measures as may be necessary to maintain the integrity of the primary containment system as long as necessary following the various postulated design basis loss-of-coolant accidents.

c. The reactor building, encompassing the primary containment system, provides secondary containment when the primary containment is closed and in service, and provides for primary containment when the primary containment system is open.

d. Provisions are made for preoperational pressure and leak rate testing of the primary containment system and for leak testing at periodic intervals. Provision is also made for leak testing selected penetrations. Provision is also made for demonstrating the functional integrity of the secondary containment system.

e. The integrity of the complete plant containment system and such other associated engineered safeguards as may be necessary are designed and maintained so that offsite doses resulting from postulated design basis accidents are below the values stated in 10CFR100.

1.2.5 Plant Instrumentation and Control

a. The plant is provided with a main control room having adequate shielding and air conditioning facilities to permit occupancy for normal plant operation as well as during all postulated design basis accident situations.

b. Interlocks or other protective devices are provided so that procedural controls are not the only means of preventing serious accidents.

c. A reliable reactor protection system, independent from the reactor process control system, is provided to automatically initiate appropriate action whenever plant conditions approach pre-established limits. Periodic testing capability is provided. Sufficient redundancy is provided so that failure or removal from service of any one component or portion of the system will not preclude appropriate actuation of the reactor protection system when required.

1.2.6 Plant Electrical Power

Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required engineered safeguards functions under all postulated design basis accident conditions.

1.2.7 Plant Radioactive Waste Disposal

a. Gaseous, liquid and solid waste disposal systems are designed so that discharge of effluents and off-site shipments will be in accordance with 10CFR20 and other applicable regulations.

b. Process and discharge streams are appropriately monitored and such features incorporated as may be necessary to maintain releases below the permissible limits of 10CFR20.

1.2.8 Plant Shielding and Access Control

The radiation shielding in the plant and the plant access control patterns are such that the personnel doses are as low as reasonably achievable and well below the limits of 10CFR20.

1.2.9 Plant Fuel Handling and Storage

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide cooling for the spent fuel.

1.2.10 Separation of Safety Systems

Systems and equipment provided for the prevention of and the mitigation of the consequences of accidents are provided in such redundancy and physical separation that the accident will not preclude operation of sufficient equipment to effectively control the effects of the accident.

1.2.11 Class I Equipment and Structures

Class I structures, systems and components are those whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant under normal or accident conditions and to the removal of decay and sensible heat from the reactor.

1.2.12 Class II Equipment and Structures

Class II structures, systems and components are those whose function is not vital or essential to safe shutdown.

1.3 Summary Design Description and Safety Analysis

1.3.1 Plant Site and Environs

Section 2 provides detailed information on the site and environs of the Monticello Nuclear Generating Plant which confirms the suitability of the site. This section summarizes the principal design characteristics of the site and environs.

1.3.1.1 Description of the Site

The plant is located within the city limits of Monticello, Minnesota (1990 population 4,941), on the south bank of the Mississippi River in Section 33, T-122N, R-25W, in Wright County, Minnesota, at 45° 20' N latitude and 93° 50' W longitude. Approximately 2150 acres of land are owned in fee by Northern States Power Company (NSP) at the plant location. NSP is a wholly owned utility operating subsidiary of Xcel Energy Corporation (Xcel Energy) (Reference 3). The property is divided by the river with part being in Sherburne County and part in Wright County. Drawing ND-95208, Section 15, shows the Monticello property map. 01-110

The immediate plant area, including major portions of the intake, is completely enclosed by a double security fence. Access to the plant is through the Security Building or Security Gate. The access road extends from the security gate to County Road 75, 3000 feet southeast of the reactor building. Interstate Highway #94 is located 3700 feet southwest. Railroad access is provided by the Burlington Northern Railroad. Air access is provided by the Twin Cities Airport of Minneapolis - St. Paul, located approximately 45 miles southeast of the site.

1.3.1.2 Description of the Environs

The area around the site is used for agriculture. The nearest house to the reactor building is about 0.6 miles southwest. The nearest well serving more than one home is located in the city of Monticello. The city, which consists of a small commercial complex and attendant residential development, includes the Wright County portion of the plant site within its boundary. The population within the 10 mile Emergency Planning Zone (EPZ) (1996 estimate) is approximately 41,950. Population density within the 5 mile EPZ (1996 estimate) is approximately 350 per square mile. The northwestern suburbs of Minneapolis are about 30 miles from the site.

From the population and land usage viewpoint, it is concluded that the site is suitable for the plant, considering the containment and additional engineered safeguards provided as an integral part thereof.

1.3.1.3 Geology

The site area is covered by unconsolidated deposits of dense soils underlain by Paleozoic sedimentary rock at about 75 to 122 feet. The reactor building foundation is founded on a layer of compacted granular backfill overlying a dense sand and gravel layer which covers a fine to medium grain sandstone. The turbine building foundation rests on a dense layer of sand and gravel on which the reactor backfill is founded. Seismic surveys disclosed no unusual or extreme subsurface conditions.

The geology of the area and soil tests indicate that the rock and soil loading capacity is adequate to support the reactor building and related structures.

1.3.1.4 Hydrology

The finished plant grade is about 25 feet above mean river level, (905 MSL), 14 feet above the record flood (916 feet MSL-1965), and 9 feet above the predicted 1,000 year flood.

The "probable maximum flood" criterion as defined by the U.S. Corps of Engineers was used to establish the maximum flood level. Using this criterion, the flood analyses predicted a probable maximum flood peak stage at the site of approximately nine feet above plant grade. The peak level at the site would be reached in about 12 days from the onset of the worst combination of hydrometeorological, hydrological and climatic conditions resulting in the probable maximum flood.

River flows vary widely throughout the year. Generally maximum flows occur in the spring, and minimum flows occur in late summer (July, August, September) or mid-winter (January, February). The low flow of record is 220 cfs. The mean flow is 3400 cfs and the average flow is 4600 cfs. The plant design and construction (including radioactive waste control systems) and contingency procedures take into consideration the extremes of river flow and stage (i.e., the probable maximum flood).

1.3.1.5 Regional and Site Meteorology

The meteorology of the site area is basically that of a continental location with its associated favorable atmospheric dilution conditions prevailing. Diffusion climatology comparisons with other locations indicate that the site is typical of the midwestern United States. Inversion conditions exist at the site approximately 30 to 40% of the time.

All Class I and II structures are designed to withstand the maximum potential loadings resulting from a wind speed of 100 mph at 30 feet above ground with a gust factor of 1.1. The design is in accordance with standard codes and normal engineering practices.

It is estimated that the probability of experiencing tornadic forces at the site is of the order of one chance per 2000 years. In spite of this low probability, the plant is designed so that features of the plant important to continuity of reactor core cooling are designed to withstand the forces of short term tornado loadings of 300 mph.

1.3.1.6 Seismology and Design Response Spectra

The seismic design for critical structures and equipment for this plant is based on dynamic analysis of acceleration or velocity response spectrum curves which are based on a horizontal ground acceleration of 0.06 g.

The natural periods of vibration are calculated for buildings and equipment which are vital to the safety of the plant. Damping factors are based upon the materials and methods of construction used.

Earthquake design is based on ordinary allowable stress as set forth in the applicable codes and is very conservative because the usual one-third increase in allowable working stresses due to loadings from the operating basis earthquake is not used. As an additional requirement, the design is such that a safe shutdown can be made following a safe shutdown earthquake assuming a horizontal ground acceleration of 0.12 g. 02-919

The 0.12 g design criteria are for critical items only; that is, for Class I items. 02-921

1.3.1.7 Environmental Monitoring Program

An environmental radiation monitoring program was initiated in 1968 prior to the start of plant operation and continued after plant operation began.

The current Radiological Environmental Monitoring Program (REMP) is a comprehensive program of sampling and analysis of the air, terrestrial, and aquatic environments for radioactivity. The types of samples and sample locations included in the current REMP at Monticello are specified in the Offsite Dose Calculation Manual (ODCM) (Reference 1).

1.3.2 Reactor System

The reactor is a single-cycle, forced circulation, low power density boiling water reactor producing steam for direct use in a steam-turbine. The reactor core includes the fuel assemblies and control rods.

The reactor core is assembled in modules of four fuel assemblies set in the interstices of a cruciform control rod. This modular core form, common to all General Electric boiling water reactors, permits substantial increase in thermal power with a small increase in core diameter and at the same time preserves the reactivity control characteristics.

The reactor vessel contains the reactor core and supporting structure, steam separator and dryer assemblies, jet pumps, control rod guide tubes, and the Reactor Feedwater, Emergency Core Cooling System (ECCS), and Standby Liquid Control System spargers and other components as shown in Figure 3.6-1. The inside diameter of the reactor vessel is approximately 17-feet 1-inch and the inside height between top head and bottom cap is approximately 63-feet 2-inches. Some of the main connections to the reactor vessel include the four main steam lines, two jet pump motive flow recirculation loop lines, four reactor feedwater lines, ten jet pump inlet lines, and one hundred and twenty-one control rod drive thimbles. Other connections are provided for the reactor Standby Liquid Control System, ECCS, and the various instrumentation and control systems.

The fuel for the reactor core consists of slightly enriched uranium dioxide pellets contained in sealed Zircaloy tubes. These fuel rods are assembled into individual fuel assemblies with either an 81 rod array or a 100 rod array. Not all positions in these arrays are occupied by fuel rods. Each fuel assembly is fitted with a Zircaloy flow channel. Water serves as both the moderator and coolant for the core. The complete core loading consists of 484 fuel assemblies. 02-107

Control of reactivity is accomplished through control rod movements. The control rods are of cruciform shape and are dispersed throughout the lattice of fuel assemblies. The control rods are of the bottom-entry type and are moved vertically within the reactor core by individual, hydraulically operated, locking piston type control rod drives. 02-958

The Control Rod Drive System is designed to allow control rod withdrawal or insertion at a limited rate, one control rod at a time, for power level control and flux shaping during reactor operation. Stored energy available from gas-charged accumulators and/or from reactor pressure provides hydraulic power for rapid simultaneous insertion of all control rods for rapid (scram) reactor shutdown. Each control rod has its own separate drive mechanism, control, and scram devices.

The operational reactivity control system is of the same design as those used in other General Electric designed reactors. Temporary control curtains fabricated of boron stainless steel were installed between fuel channels during early life of the initial core to supplement the reactivity control of the control rods.

Reactor coolant enters the bottom of the reactor core and flows upward through the fuel assemblies where boiling produces steam. The steam-water mixture is separated by steam separator and dryer assemblies located within the reactor vessel. The steam passes through main steam lines to the turbine. The separated water mixes with the incoming feedwater and is returned to the core bottom inlet through jet pumps located within the reactor vessel. The motive force for the jet pumps is supplied by the water from the two Reactor Recirculation System loops. Each loop has a variable speed centrifugal pump with mechanical seals, motor operated gate valves for isolation of pumps for maintenance, and instrumentation for recirculation flow measurements and control.

Heat balances for the reactor system are shown on Figure 1.3-1.

1.3.3 Plant Containment System

The Primary Containment System, consisting of a steel lightbulb-shaped drywell, a steel doughnut-shaped pressure suppression chamber, and interconnecting vent pipes, provides the first containment barrier surrounding the reactor vessel and reactor primary system. Any leakage from the Primary Containment System is to the Secondary Containment System which consists of the reactor building, the plant Standby Gas Treatment System, and the plant main stack. The integrated plant containment system and its associated engineered safety features are designed so that off-site doses resulting from postulated design basis accidents are well below the reference values stated in 10CFR100.

1.3.3.1 Primary Containment System

The primary containment is designed to accommodate the pressures, temperatures, and hydrodynamic loads which would result from, or occur subsequent to a postulated loss-of-coolant accident (LOCA) within the primary containment and safety relief valve operations. The LOCA conditions evaluated in the primary containment design include the Design Basis Accident (DBA); that is, the pipe failure equivalent to a double-ended, circumferential rupture of a Reactor Recirculation System line resulting in the loss of reactor water at the maximum rate. The pressure suppression chamber is a steel, torus-shaped pressure vessel approximately half filled with water, and located below and encircling the drywell. The vent system from the drywell terminates below the water level of the pressure suppression chamber so that in the event of any pipe failure in the drywell, the released steam would pass directly to the water where it would be condensed. A bellows assembly connecting the suppression chamber to the vent line allows for differential movement between the drywell and the suppression chamber.

Isolation valves are provided on piping, penetrating the drywell and the suppression chamber, to provide integrity of the containment when required. These valves are actuated automatically by signals received from the containment isolation system. The valves on the auxiliary process systems are left open, or are closed, depending upon the functional requirements of that system, without reducing the integrity of the primary containment system.

Two features are included in the primary containment design to aid in maintaining the integrity of the Primary Containment System in the event of a postulated design basis loss-of-coolant accident. The containment spray cooling mode of the Residual Heat Removal System (RHR) provides redundant cooling capability for the removal of heat within the drywell and the pressure suppression chamber. Capability is provided in the containment structure design to withstand the forces exerted in the event that it is necessary to flood the primary containment vessel (drywell and suppression chamber) to a level which would flood the reactor core.

Each safety relief valve discharge line is routed to a standard Mark I T-quencher discharge device located in the suppression chamber below the water level. The T-quencher discharge device is used to ensure stable steam condensation at expected pool temperatures, and to mitigate pressure, thrust and hydrodynamic loads on the safety relief valve discharge line piping and the suppression chamber resulting from safety relief actuations.

After complete installation of all penetrations in the drywell and suppression chamber, these vessels were pressurized to the design pressure and measurements taken to verify that the integrated leakage rate from the integral vessels did not exceed the design leakage rate.

All containment closures which are fitted with resilient seals or gaskets are separately testable at pressures up to and including the containment design pressure to verify leak tightness. The covers on flanged closures, such as the equipment access hatch cover, the drywell head and access manholes are provided with double seals and with a test tap which allows pressurizing the space between the seals without pressurizing the entire primary containment system. Similarly, the space between the dual air lock doors can be pressurized to full design pressure.

Electrical penetrations have been provided with double seals and can be separately tested at pressures up to and including the containment design pressure. The test taps and the seals are located so that the tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber.

Those pipe penetrations which must accommodate thermal movement are provided with expansion bellows-type seals. The bellows expansion joints are designed for the primary containment system design pressure and can be checked for leak tightness when the Primary Containment System is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized up to and including the containment design pressure to permit testing the individual penetrations for leakage.

1.3.3.2 Secondary Containment System

Secondary containment is a controlled volume within the Reactor Building. The primary safeguards functions of the secondary containment are to minimize ground level release of airborne radioactive materials, and to provide for controlled, filtered, elevated release of the secondary containment atmosphere under postulated design basis accident conditions. Most of the Reactor Building is part of the secondary containment, and the Reactor Building provides the structural integrity of the secondary containment.

A plant Standby Gas Treatment System is provided to filter the secondary containment ventilation exhaust and discharge it to the off-gas vent stack during plant secondary containment system isolation conditions.

1.3.4 Plant Auxiliary and Standby Cooling Systems

In addition to the turbine-generator and the main condenser systems, multiple, independent, auxiliary process systems are provided for the purpose of cooling the reactor and primary containment system under various normal and abnormal conditions.

a. A Reactor Core Isolation Cooling System (RCIC) is provided for a continuous supply of makeup and cooling water to the reactor core when the reactor is isolated from the turbine and when the normal feedwater systems are not available.

b. A two loop Core Spray System (CS) is provided. The system is designed to pump water from the suppression chamber pool directly to the reactor core through spargers mounted in the reactor vessel above the active core in a manner which will prevent fuel clad damage after depressurization following postulated design basis loss-of-coolant accident.

c. A residual heat removal system (RHR) is provided which serves the following functions:

   (1) To inject water into the reactor vessel after depressurization following a postulated design basis loss-of-coolant accident in order to rapidly reflood the core and prevent fuel clad melting. (This is the Low Pressure Coolant Injection System (LPCI) mode of RHR.)

   (2) To remove heat from the water in the suppression chamber pool. (This is the containment cooling mode of RHR.)

   (3) To spray water into the drywell and torus as an augmented means of removing energy from the drywell as required subsequent to a postulated design basis loss-of-coolant accident. (This is the containment spray mode of RHR.)

   (4) To remove decay heat and sensible heat from the reactor primary system so that the reactor can be shut down for a normal refueling and service operation. (This is the reactor shutdown cooling and head spray cooling mode of RHR.)

d. A High Pressure Coolant Injection System (HPCI) is provided for removal of decay heat and to provide coolant inventory control and heat dissipation from the core to the suppression chamber to prevent fuel clad damage following a postulated small break loss-of-coolant accident.

e. An Automatic Depressurization System (ADS) is provided which, together with the core spray or the LPCI mode of the RHR will prevent fuel clad damage following a postulated small break loss-of-coolant accident.

f. A standby coolant supply system is provided by a cross-tie between the Plant Service Water System and the Feedwater System which makes available an inexhaustible supply of cooling water from the river to the reactor core and containment independent of all other cooling water sources.

g. An intertie is provided between the RHR Service Water System and the RHR discharge line. The plant Fire Protection System is also capable of discharging into this intertie line. This intertie provides an inexhaustible source of river water to the reactor core.

The core cooling provisions itemized above (except f.) are designed to prevent fuel clad damage for the full range of primary system pipe size ruptures which may be postulated to occur without reliance upon offsite sources of power.

1.3.5 Plant Instrumentation Control Systems

1.3.5.1 Reactor Control

Reactor power is controlled by movement of control rods and by regulation of the reactor coolant recirculation system flow rate. Control rods are also used to shape the reactor core power distribution. Procedural controls backed up by protective devices (reactor protection system, etc.) are used so that reactor core thermal performance does not exceed pre-established limits.

Reactor steam flow is automatically controlled by the Main Steam Pressure Control System which adjusts and controls steam flow to the main turbine in response to turbine inlet pressure. As a result, the plant turbine-generator power output follows the reactor power output.

A main turbine bypass system, having a capacity of approximately 14% of steam flow at rated load, is supplied with the turbine-generator system to restrict overpressure transients resulting from sudden complete or partial closure of the main turbine control valves or stop valves and provides a means of releasing steam to the main condenser during shutdown operations. The main turbine bypass system valves are operated on an overpressure signal from the Main Steam Pressure Control System. Rapid partial load rejection (up to 14% of rated flow) can be accommodated with the main turbine bypass system.

1.3.5.2 Reactor Protection System

A Reactor Protection System is provided which automatically shuts down the reactor whenever the plant parameters monitored by the system approach pre-established limits.

The Reactor Protection System consists of two separately powered trip systems, Channel A and Channel B, each made up of two subchannels. The protection system receives inputs from sensors monitoring plant parameters. Each subchannel receives an input from at least one independent sensor monitoring each of the parameters. An unbypassed trip occurring in either subchannel (or both) of logic channel A, together with a unbypassed trip occurring in either sub-channel (or both) of logic channel B results in the opening of the scram valves in the Control Rod Drive System causing rapid insertion (scram) of the control rods. The Reactor Protection System is designed to cause a scram on loss of power to the Reactor Protection System. Components of the Reactor Protection System can be removed from service for testing and maintenance without interrupting plant operations and without negating the ability of the Reactor Protection System to perform its protective functions upon receipt of appropriate signals.

1.3.5.3 Plant Radiation Monitoring System

Instrumentation is provided for continuous monitoring of the radioactivity of specified process systems. Process systems where significant amounts of radioactivity may be present are monitored for any variation from normal. Certain nonradioactive processes are monitored to provide an alarm in the event they become contaminated due to the failure of a radiation barrier.

1.3.6 Plant Fuel Storage and Handling Systems

The refueling procedure is generally referred to as "wet" refueling since all irradiated fuel is always kept under water. The wet refueling procedure allows visual control of operations at all times. This feature is instrumental in producing a safe, efficient refueling sequence.

Spent fuel discharged from the reactor is transferred under water through the spent fuel storage pool canal into storage racks provided in the storage pool. The spent fuel storage pool is designed to accommodate the channel stripping operation and the many other fuel maintenance and inspection operations that are required. The spent fuel racks are designed and arranged so that the risk of criticality is eliminated. Storage space is also provided in the pool for irradiated fuel assembly channels and replaced control rods, the spent fuel shipping cask, and the certain small internal components of the reactor core.

New fuel is brought through the equipment entrance of the Reactor Building and hoisted to the upper floor utilizing the reactor building crane. The new fuel may be stored in the new fuel vault located adjacent to the spent fuel storage area within the Reactor Building or may be placed directly in the spent fuel storage racks prior to insertions into the reactor.

Monticello has elected to comply with the criticality accident requirements of 10CFR50.68 in lieu of 10CFR70.24 for the handling and temporary storage of new fuel and non-fuel special nuclear material. Plant equipment and procedures have been evaluated and found to meet the requirements specified in 10CFR50.68 for precluding criticality events outside of the reactor.

1.3.7 Plant Main Turbine System

The saturated steam leaving the reactor vessel flows through the four main steam lines to the main turbine located in the turbine building. After passing through the main turbine, the low pressure steam is condensed, the noncondensible gases are removed, and the condensate is demineralized before being returned to the reactor vessel through the Reactor Feedwater System heaters. Heat balances for the plant turbine system including the extraction steam subsystem are shown on Figure 1.3-2. A simplified process flow diagram for the entire plant is shown on Figure 1.3-3.

1.3.8 Plant Electrical Power Systems

The electrical output of the plant is fed into the plant site high voltage switchyard, and from the yard to Xcel Energy's network grid system via independent transmission lines (two 345 KV, two 230 KV and three 115 KV transmission lines). Plant Auxiliary electrical power is supplied from the 115 KV lines, and/or from the 345 KV switchyard. The plant Emergency Diesel Generator System (2 essential and 1 non-essential units) provides onsite standby emergency auxiliary electrical power. 01-010

The plant DC battery system consists of two 125 Vdc and three 250 Vdc batteries and systems which provide for controls and instrumentation which are vital to reactor and overall plant safety and to power certain functional requirements for reactor shutdown. Two separate 24 Vdc battery systems supply the Nuclear Instrumentation System and process radiation monitoring system.

1.3.9 Plant Shielding, Access Control, and Radiation Protection Procedures

Control of radiation exposure of plant personnel and people external to the plant is accomplished by a combination of radiation shielding, control of access into certain areas, and administrative procedures. The requirements of 10CFR20 were used as a basis for establishing the basic criteria and design bases.

Shielding is used to reduce radiation dose rates in various parts of the plant to acceptable limits consistent with operational and maintenance requirements. Access control and administrative procedures are used to limit the integrated dose received by plant personnel as low as reasonably achievable (ALARA). Access control and procedures are also used to limit the potential spread of contamination from various areas, particularly areas where maintenance occurs. The table below summarizes the design bases for shielding to assure that radiation levels in various areas of the plant are consistent with operational requirements.

Design Bases Plant Shielding Requirements

Degree of Access Required	Design Dose Rate (DDE, mrem/hr)
a) Continuous Occupancy	
Outside Controlled Access Area	0.5
Inside Controlled Access Area	1
b) Occupancy to 10 hr/week	6
c) Occupancy to 5 hr/week	12

The above design bases are at the shield walls. Generally, areas away from a shield wall receive lesser dose rates and this plus occupancy factors reduces the integrated dose received. Personnel involved in all phases of operation and maintenance normally receive far less than the permissible dose.

Both operating and shutdown conditions were considered in establishing the shielding design.

Shielding is also used as necessary to protect equipment from radiation damage. Of principal concern are organic materials such as insulation, linings, and gaskets.

1.3.10 Plant Radioactive Waste Control Systems

A Gaseous Radwaste System is provided to control, recombine, filter, store, monitor, and record the process off-gases as appropriate before release through the main plant stack during normal and abnormal plant operation.

A Liquid Radwaste System is provided for control, collection, treatment, storage, and disposal of liquid wastes. Liquid wastes are collected in sumps and drain tanks and transferred to the radwaste facility for further treatment, storage, or disposal.

In the Liquid Radwaste System, liquid wastes to be discharged from the system are handled on a batch basis with each batch being analyzed and disposed of as required. The system is designed to permit the waste to either be returned to the plant condensate system or be released to the Mississippi River after dilution in the discharge canal.

A Solid Radwaste System functions so that solid wastes are treated, sorted, packaged, solidified (if necessary), and shipped offsite.

1.3.11 Summary Evaluation of Plant Safety

1.3.11.1 General

The general safeguards objectives of the design of this plant are to protect the equipment and to prevent radiation exposures in excess of a small fraction of pre-established limits to any persons on or off the plant premises, either during normal operation or during credible or postulated design basis accident conditions.

In order to meet these objectives, the design and operation include the following:

a. Means for positive control of plant process parameters important to safety.

b. Inherent safety features and automatic devices are included in the design to prevent a reactor operator error or equipment malfunction from causing an accident. Tests are conducted periodically to assure proper functioning of such devices.

c. Multiple barriers are provided to contain the radioactive materials. The reactor core is conservatively designed to operate with thermal parameters significantly below those which could lead to fuel damage.

d. The plant operating personnel are thoroughly knowledgeable in the operating characteristics of the plant, and are trained to follow written procedures to minimize the occurrence of operating errors.

1.3.11.2 Summary of Offsite Doses

The plant radioactive waste control systems for normal operation are designed to limit the radiation exposure of the offsite neighboring population to within the design objectives of Appendix I to 10CFR50.

1.4 Identification and Qualification of Contractors

1.4.1 Licensee

The Monticello Nuclear Generating Plant (MNGP) is owned by Northern States Power Company (NSP), a wholly owned utility operating subsidiary of Xcel Energy Corporation (Reference 3). Nuclear Management Company, LLC (NMC) succeeds NSP as the operator of the plant. Consequently, NMC is authorized to act as agent for NSP and has exclusive responsibility and control over physical construction, operation, and maintenance of the facility (Reference 4). 01-005 01-010

1.4.2 Contractors

The Monticello Nuclear Generating Plant, Unit 1, was designed and built by General Electric Company as prime contractor for Northern States Power Company. General Electric engaged the services of the Bechtel Corporation as architect-engineer to provide the non-nuclear design and as engineer constructor for the plant.

Preoperational testing of equipment and systems and initial operation were performed by Northern States Power Company personnel with the technical assistance of General Electric. The initial staff for the Monticello Plant was drawn largely from the experienced staff of the Pathfinder Atomic Power Plant which Northern States Power Company operated. In addition, the initial staff had undergone extensive training during the construction phase of the plant. The plant was turned over to Northern States Power Company and responsibility for operation was assumed by NSP following demonstration of the operational capability at the contract specified output.

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy, entered into a Nuclear Power Plant Operating Services Agreement with NMC. In accordance with that contract, NMC has assumed exclusive responsibility for the operation and maintenance of the Monticello Nuclear Generating Plant (Reference 4). 01-005

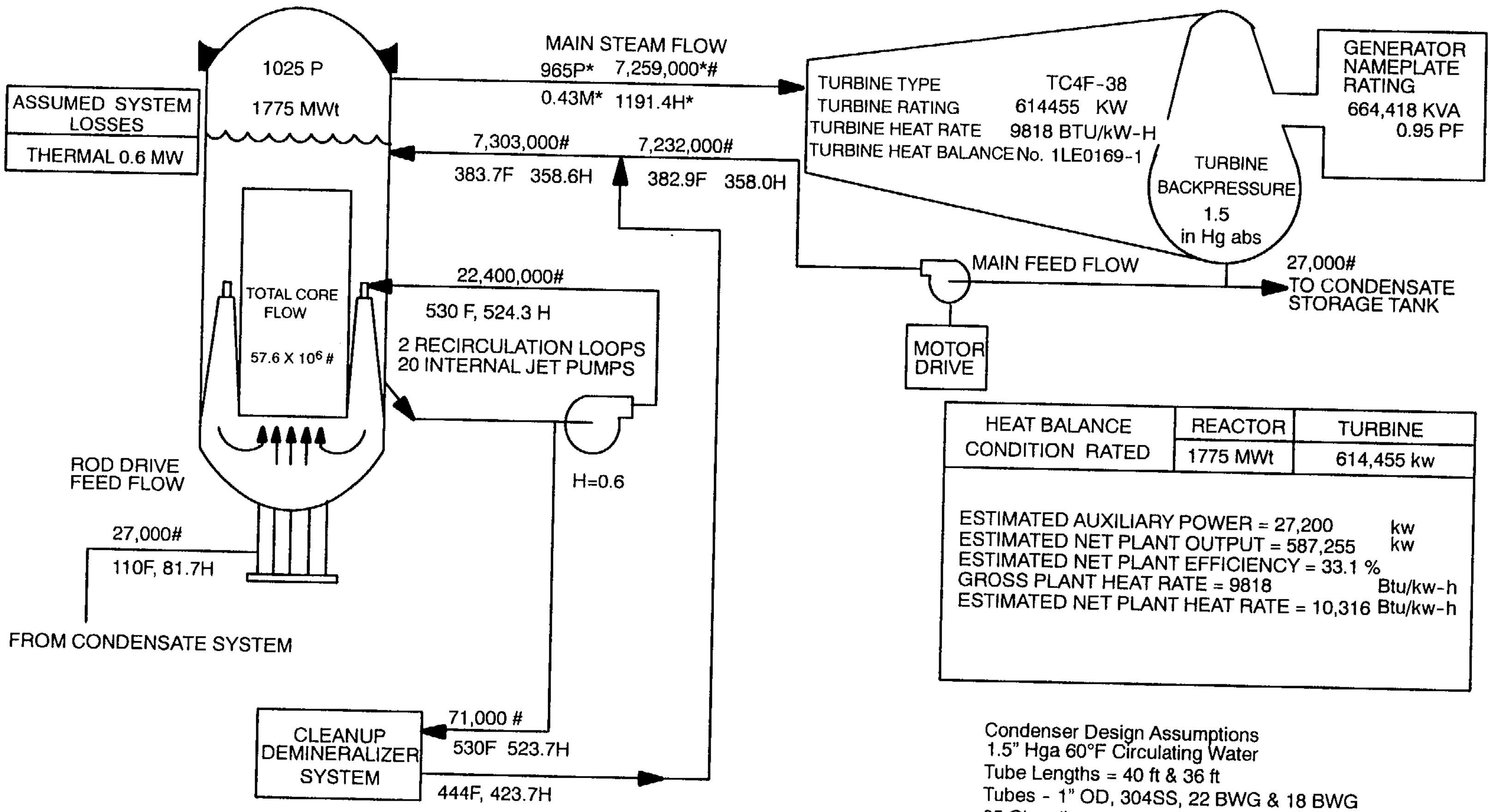
1.5 References

1. Monticello Nuclear Generating Plant, Offsite Dose Calculation Manual (ODCM).
2. NSP (M F Hammer) letter to the NRC, "Power Rerate Test Program Startup Report," dated February 18, 1999.
3. NRC (C F Lyon) letter to NSP (W H Brunetti), "Issuance of Conforming Amendment re: Transfer of Facility Operating License from Northern States Power Company to a New Utility Operating Company Subsidiary (TAC No. MA7003)," dated August 18, 2000. 01-010
4. NRC (C F Lyon) letter to NMC (M B Sellman), "Issuance of Conforming Amendment re: Transfer of Operating Authority Under the Facility Operating License from Northern States Power Company to Nuclear Management Company, LLC (TAC No. MA7313)," dated August 7, 2000. 01-005

FIGURES

I/djm

Figure 1.3-1 Nuclear Steam Supply Heat Balance 1775 MWt (631.2 MWe)



Condenser Design Assumptions
1.5" Hga 60°F Circulating Water
Tube Lengths = 40 ft & 36 ft
Tubes - 1" OD, 304SS, 22 BWG & 18 BWG
85 Cleanliness Factor
Single Pass Unit
Pump Head = 27.8 ft

Legend
\# - Flow, lb/Hr
P - Pressure Psia
H - Enthalpy, Btu/lb
M - Moisture Percent
F - Temperature, °F

Note: Some values are representative nominal values.

*THESE REPRESENT TURBINE INLET CONDITIONS

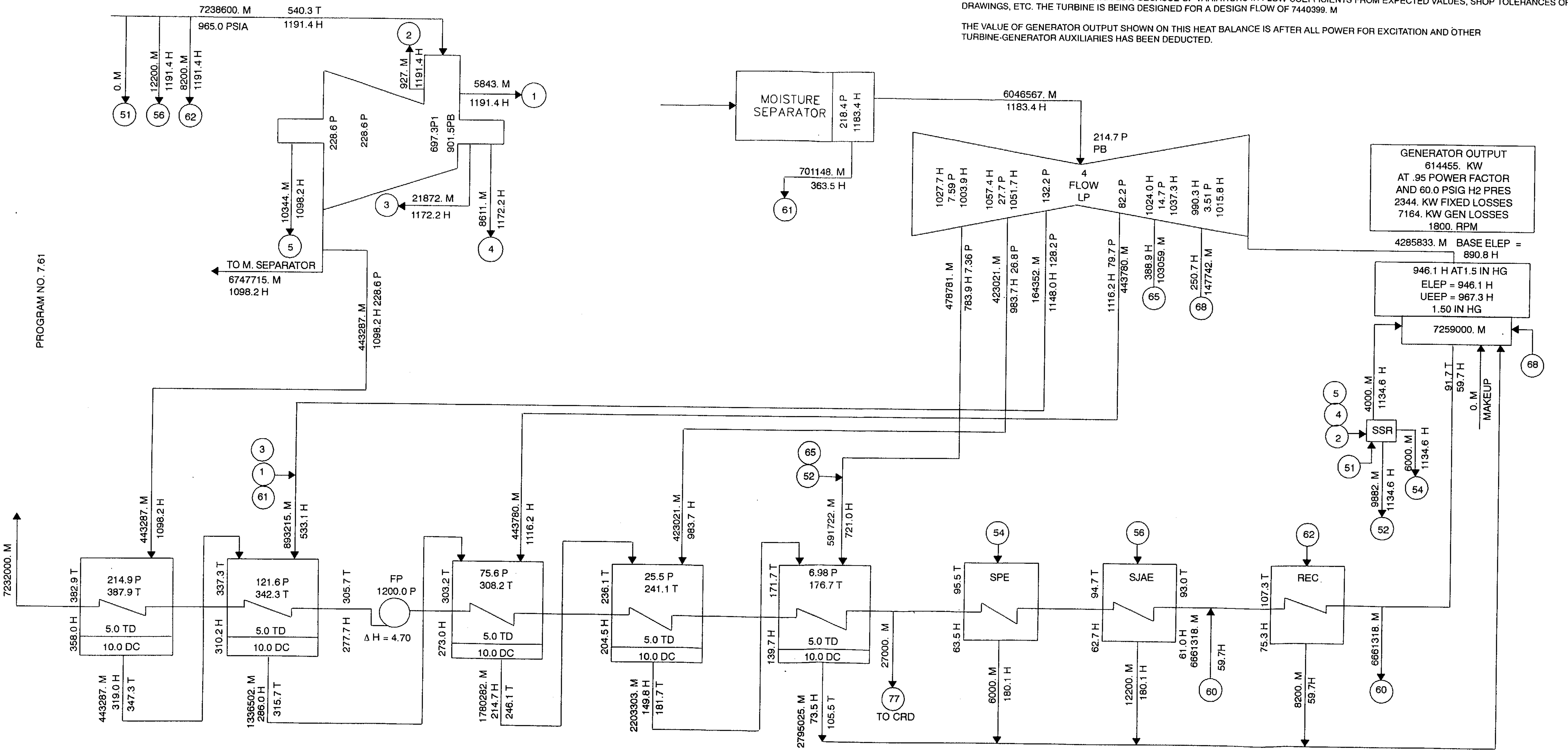
Figure 1.3-2 Turbine Generator System - Heat Balance 1775 MWt (631.2 MWe)

TURBINE AND EXTRACTION ARRANGEMENT IS SCHEMATIC ONLY - CALCULATED DATA - NOT GUARANTEED

RATING FLOW IS 7223689. M AT INLET STEAM CONDITIONS OF 965.0 PSIA AND 1191.4 H TO MINIMIZE THE POSSIBILITY THAT THE TURBINE WILL BE UNABLE TO PASS RATING FLOW BECAUSE OF VARIATIONS IN FLOW COEFFICIENTS FROM EXPECTED VALUES, SHOP TOLERANCES OF DRAWINGS, ETC. THE TURBINE IS BEING DESIGNED FOR A DESIGN FLOW OF 7440399. M

THE VALUE OF GENERATOR OUTPUT SHOWN ON THIS HEAT BALANCE IS AFTER ALL POWER FOR EXCITATION AND OTHER TURBINE-GENERATOR AUXILIARIES HAS BEEN DEDUCTED.

PROGRAM NO. 7.61



VALVE BEST POINT GROSS HEAT RATE = $\frac{7238600. (1191.4 - 358.0)}{614455.}$ = 9818 $\frac{BTU}{KW\text{-}HR}$

LEGEND - CALCULATIONS BASED ON 1967 ASME STEAM TABLES
M - FLOW - LB/HR
P - PRESSURE - PSIA
H - ENTHALPY - BTU/LB
T - TEMPERATURE - F DEGREES

613240. KW 1.50 IN HG ABS
TC4F 38.0 IN LSB 1800 RPM
965.0 PSIA 1191.4 BTU/LB NON REHEAT
GEN - 664400.KVA .95 PF LIQ 60.0 PSIG H2 PRES

B.6.1-06-2

I/djm

Figure 1.3-3 Plant Process Flow Diagram

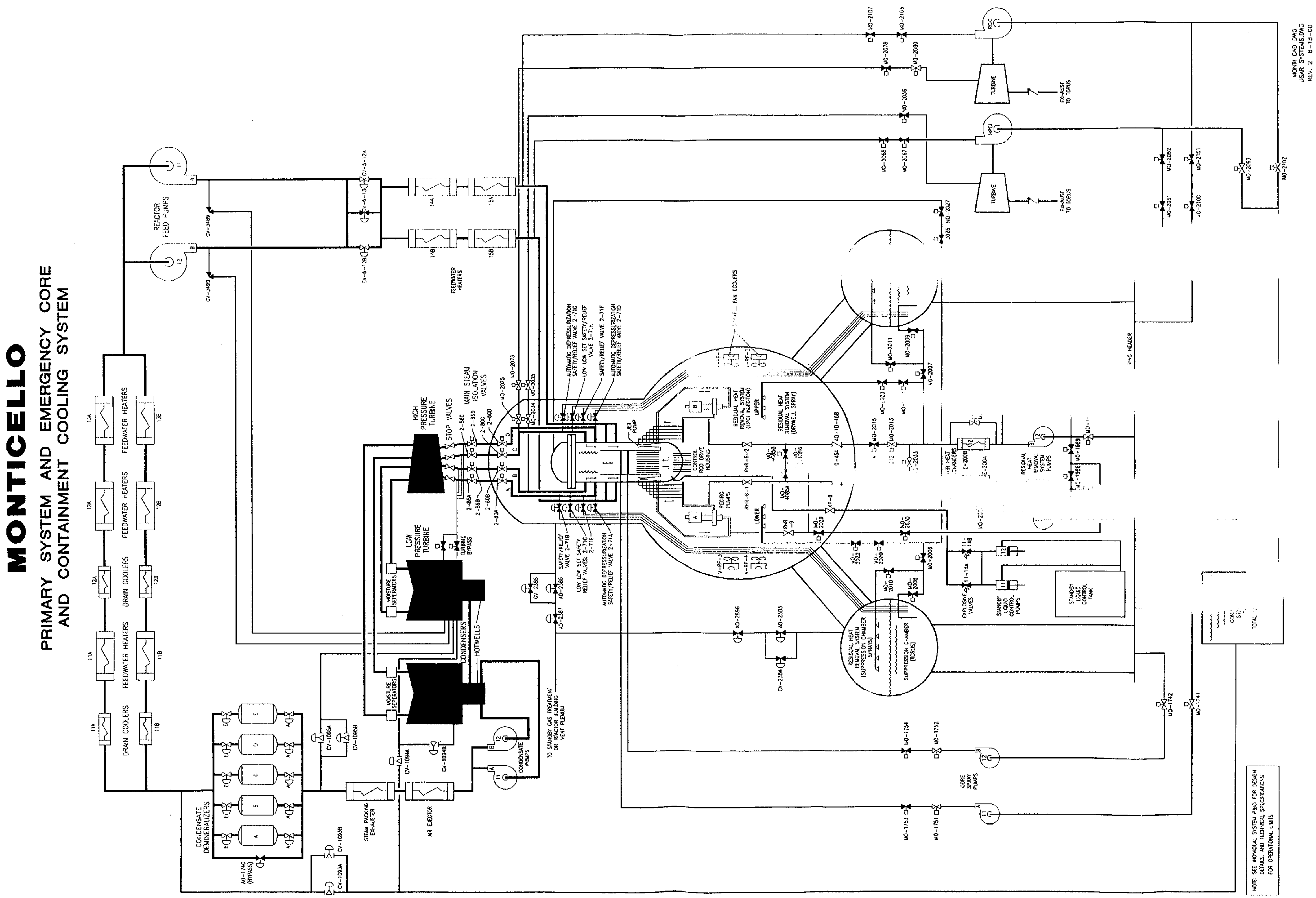


TABLE OF CONTENTS

Section		Page
2.1	**Introduction**	1
2.2	**Site Description**	1
2.2.1	Location	1
2.2.2	Topography	1
2.2.3	Access	2
2.2.4	Land Use	2
2.2.5	Population Distribution	2
2.2.6	Conclusions	3
Table 3.2-1	Estimated 1990 Population Distribution Around the Monticello Nuclear Generating Plant	4
Table 3.2-2	Estimated 2000 Population Distribution Around the Monticello Nuclear Generating Plant	5
2.3	**Meteorology**	1
2.3.1	General	1
2.3.2	Temperature	1
2.3.3	Precipitation	1
2.3.4	Winds and Wind Loading	2
2.3.4.1	Tornadoes and Severe Thunderstorms	3
2.3.4.2	Conclusions	3
2.3.5	Plant Design Based on Meteorology	4
Table 2.3-1	Monthly Air Temperature	5
Table 2.3-2	Summary of Precipitation Statistics	6
Table 2.3-3	Average Monthly Snowfall (inches)	7
Table 2.3-4	Snow Load Data	8
Table 2.3-5	Wind Frequency Distributions at 10 Meter Level, Stability Class A	9
Table 2.3-6	Wind Frequency Distributions at 10 Meter Level, Stability Class B	10
Table 2.3-7	Wind Frequency Distributions at 10 Meter Level, Stability Class C	11

Section		Page
Table 2.3-8	Wind Frequency Distributions at 10 Meter Level, Stability Class D	12
Table 2.3-9	Wind Frequency Distributions at 10 Meter Level, Stability Class E	13
Table 2.3-10	Wind Frequency Distributions at 10 Meter Level, Stability Class F	14
Table 2.3-11	Wind Frequency Distributions at 10 Meter Level, Stability Class G	15
Table 2.3-12	Wind Frequency Distributions at 10 Meter Level, All Classes Combined	16
Table 2.3-13	Wind Frequency Distributions at 100 Meter Level, Stability Class A	18
Table 2.3-14	Wind Frequency Distributions at 100 Meter Level, Stability Class B	19
Table 2.3-15	Wind Frequency Distributions at 100 Meter Level, Stability Class C	20
Table 2.3-16	Wind Frequency Distributions at 100 Meter Level, Stability Class D	21
Table 2.3-17	Wind Frequency Distributions at 100 Meter Level, Stability Class E	22
Table 2.3-18	Wind Frequency Distributions at 100 Meter Level, Stability Class F	23
Table 2.3-19	Wind Frequency Distributions at 100 Meter Level, Stability Class G	24
Table 2.3-20	Wind Frequency Distributions at 100 Meter Level, All Classes Combined	25
Table 2.3-21	Maximum Wind Velocity	27
Table 2.3-22	Annual Average Dispersion Factor (X/Q) - Reactor Building Vent Releases	28
Table 2.3-23	Annual Average Dispersion Factor (X/Q) - Plant Stack Releases	29
Table 2.3-24	Relative Deposition per Unit Area (D/Q) - Reactor Building Vent Releases	30
Table 2.3-25	Relative Deposition per Unit Area (D/Q) - Plant Stack Releases	31
Table 2.3-26	Site Boundary X/Q and D/Q - Reactor Building Vent Releases	32
Table 2.3-27	Site Boundary X/Q and D/Q -Plant Stack Releases	33

Section		Page
2.4	**Hydrology**	**1**
2.4.1	Surface Water	1
2.4.2	Public Water Supplies	2
2.4.2.1	Surface Water	2
2.4.2.2	Ground Water	3
2.4.3	Plant Design Bases Dependent on Hydrology	4
2.4.4	Water Use Permits and Appropriations Relevant to Plant Operation	4
2.4.5	Surface Water Quality	4
Table 2.4-1	Mississippi River Flows at Elk River and St. Cloud, Minnesota	6
2.5	**Geology and Soil Investigation**	**1**
2.5.1	General	1
2.5.2	Regional Geology	1
2.5.3	Site Geology	2
2.5.4	Groundwater	3
2.5.5	Foundation Investigation	3
2.5.6	Conclusions	4
Table 2.5-1	Geologic Formations in the General Area of the Site	5
2.6	**Seismology**	**1**
2.6.1	General	1
2.6.2	Seismic History	1
2.6.3	Faulting in Area	1
2.6.4	Design Criteria	2
2.6.5	Seismic Monitoring System	2
Table 2.6-1	Seismic History of the Region	4
Table 2.6-2	Modified Mercalli Intensity Scale of 1931 (Abridged)	6
2.7	**Radiation Environmental Monitoring Program (REMP)**	**1**
2.7.1	Program Design and Data Interpretation	1
2.7.2	Program Description	1
2.7.3	Interlaboratory Comparison Program	2
2.8	**Ecological and Biological Studies**	**1**

Section		Page
2.9	**Consequences of Hypothetical Local Catastrophes**	**1**
2.9.1	Toxic Chemical Spills	1
2.10	**References**	**1**
2.FIGURES		**1**
2.3-1	Return Period of Extreme Short-Interval Rainfall, Minneapolis, MN	2
2.4-1	Flow Duration Curve, Mississippi River at St. Cloud, MN	3
2.4-2	1965 Spring Flood at Monticello Site	4
2.4-3	Flood Frequency Study - Mississippi River at Monticello Site	5
2.5-1a	Overlay Regional Tectonic Map	6
2.5-1b	Regional Geology Map	7
2.5-2	Location of Original Borings	8
2.5-3	Geologic Cross Section A-A	9
2.5-4	Log of Borings Sheet 1	10
2.5-5	Log of Borings Sheet 2	11
2.6-1	Principal Earthquakes - Minnesota Region	12
2.6-2	Tectonic Map of Minnesota Region	13
2.6-3	Seismic Regionalization U.S.A.	14
2.6-4	Seismic Probability Map of U.S.A.	15
2.6-5	Seismic Response Spectra	16

2.1 Introduction

The Monticello site was thoroughly investigated as a site for a nuclear power plant and found to be suitable as evidenced by issuance of a construction permit (Docket No. 50-263) on June 19, 1967.

Section 2 contains information on the site and environs of the Monticello Nuclear Generating Station.

2.2 Site Description

2.2.1 Location

The plant is located within the city limits of Monticello, Minnesota (1990 population 4,941), on the right bank of the Mississippi River in Section 33, T-122N, R-25W, in Wright County, Minnesota, at 45° 20' N latitude and 93° 50' W longitude.

The plant site consists of approximately 2150 acres of land owned in fee by Northern States Power Company, a wholly owned operating subsidiary of Xcel Energy Corporation (Xcel Energy). Part of this property is on the left bank of the river in Sherburne County and part is on the right bank in Wright County. Drawing ND-95208, Section 15, shows the plant site boundaries. This figure also shows an outline of the minimum fenced area which defines the restricted area boundary or site boundary for gaseous releases in accordance with10CFR20 and Appendix I to 10CFR50. The exclusion zone has been arbitrarily selected to occupy the same fenced area. This more than satisfies the 10CFR100 definition of an exclusion zone. Access to the exclusion zone is restricted by a perimeter fence with No Trespassing signs posted at intervals along the fence. Access to the exclusion zone by water is not restricted by a fence; however, No Trespassing signs are placed at intervals along the shoreline of the river. 01-010

The plant is located so that the nearest property boundary is about 1600 ft to the south of the reactor building. The distance to the nearest residence is about 0.6 mile to the southwest, and the nearest large city, St. Cloud, is 22 miles upstream from the plant site. The northwestern suburbs of Minneapolis are about 30 miles southeast from the site.

2.2.2 Topography

The topography of the Monticello site is characterized by relatively level bluffs which rise sharply above the river. Three distinct bluffs exist at the plant site at elevations 920, 930, and 940 feet above mean sea level (ft msl). Normal river is 905 ft msl, and the maximum reported flood is at 916 ft msl.

Bluffs located about 1 mile north and south of the site rise to 950 ft msl. Beyond 1 mile north, the terrain is relatively level with numerous lakes and wooded areas. To the south, west, and east, the terrain is hilly and dotted with numerous small lakes.

2.2.3 Access

Highway access is available to Wright County Road 75 which is about 3000 feet southeast of the reactor building. Interstate 94 runs northwest from Minneapolis about 3700 feet southwest of the site. Drawing ND-95208, Section 15, shows the location of these highways.

Railroad access is available from the Burlington Northern track which is about 2300 feet southwest. The site is served by a spur from this line.

The reach of the Mississippi River near the site is not suitable for navigation because its gradient is very steep and numerous shoals exist due to the current.

2.2.4 Land Use

The land surrounding the site is predominantly rural. There are a few small villages and many lakes within a 15-mile radius of the site. The terrain is heavily wooded along the river, while the bluffs away from the river are cultivated and used for dairy farming. Crops raised in the area include soybeans, corn, oats, hay, and potatoes.

2.2.5 Population Distribution

The area in which the Monticello Plant is located is principally rural in character and the land is used primarily for farming. The main residential and business district of Monticello is about 3 miles southeast of the plant. Other nearby communities include: Becker (1990 population 902) about 4 miles northwest; Big Lake (1990 population of 3,113) about 5 miles east; Maple Lake (1990 population of 1,394) about 10 miles southwest; and Buffalo (1990 population of 6,856) about 10 miles south. The closest large cities are St. Cloud (1990 population 48,812) about 20 miles northwest and Minneapolis-St. Paul area (1990 population 2,407,090) about 30 miles southeast of the plant.

The population within a 10 mile radius (300 square miles) of the plant in 1990 is estimated to be 29,432. Similarly, within a 50-mile radius of the plant (approx. 7,850 square miles) the population in 1990 is estimated to be 2,273,213, of which about 90% reside in the Minneapolis-St. Paul metropolitan area. The projected population within the 50-mile radius in the year 2000 is approximately 2.25 million.

In Wright County and in Sherburne County, immediately across the Mississippi River to the Northeast, about 80% of the land is used for farming. It is expected that these two counties will remain largely agricultural.

Table 2.2-1 shows the 1990 population and Table 2.2-2 shows the estimated 2000 population, as projected.

The low population zone radius for the Monticello facility has arbitrarily been selected as one mile. Due to the sparse population of the area there will be no difficulty in taking appropriate protective action in the event of a serious accident. Based on the 10CFR100 definition of a low population zone radius and the radiological effects presented in Section 14, the selection of a one-mile radius is more than adequate.

In December of 1997, an updated Monticello site specific area evacuation study (Reference 15) was completed to assist in emergency planning. This study was based on the most recent (1996) census estimates and considered factors such as transient and seasonal population changes, special facilities, and changes in the area transportation (roadway) network.

2.2.6 Conclusions

The population distribution around the site is quite low. Good isolation from population centers is evident. Land use is devoted to agriculture. Therefore, from the population distribution and land usage viewpoint, the site is suitable for the facility as designed. The analyses of design basis accidents in Section 14 verify that maximum expected doses at or beyond the exclusion area boundary are well below the reference doses given in 10CFR100.

Table 2.2-1 Estimated 1990 Population Distribution Around the Monticello Nuclear Generating Plant

Radius (MILES)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
0-1	4	8	0	0	0	0	0	6	6	13	13	13	6	0	0	0	69
1-2	58	58	54	46	50	50	54	54	54	54	54	54	54	26	42	54	816
2-3	96	96	96	95	96	96	96	96	96	96	96	96	96	96	95	96	1,535
3-4	134	134	134	136	134	134	137	137	137	137	137	137	137	137	133	134	1,369
4-5	171	171	171	171	171	170	171	171	171	171	171	171	171	171	171	171	2,735
5-10	1421	1421	1367	1421	1421	1441	1381	1140	1339	1436	1311	1360	1360	1448	1421	1421	22,109
10-20	5071	4510	5629	8975	17947	43619	69505	41813	5737	5737	5778	5400	5090	5131	5393	5109	240,444
20-30	6872	3275	5277	12184	54892	114477	171280	157183	19948	9211	7744	4054	7206	8386	8124	7422	597,535
30-40	5590	4315	6916	8069	73132	190714	239887	91658	18844	9382	6310	4531	8345	11759	11461	7582	698,495
40-50	5036	4845	5460	11932	33812	299783	233190	36340	15140	9304	7761	6223	10319	15300	8351	4511	707,307
TOTAL	24,453	18,833	25,104	43,028	181,655	650,485	715,701	328,598	61,472	35,541	29,375	22,039	32,784	42,454	35,191	26,500	2,273,213

Table 2.2-2 Estimated 2000 Population Distribution Around the Monticello Nuclear Generating Plant

Radius (MILES)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
0-1	6	12	0	0	0	0	0	9	9	19	19	19	9	0	0	0	102
1-2	85	85	79	66	73	73	79	79	79	79	79	79	79	38	61	79	1,192
2-3	139	139	139	139	139	138	139	139	139	139	139	139	139	139	138	139	2,222
3-4	194	194	194	194	194	196	199	199	199	199	199	199	199	199	192	194	3,144
4-5	249	249	249	249	249	247	249	249	249	249	249	249	249	249	248	249	3,981
5-10	2062	2062	1983	2062	2062	2095	2008	1657	1948	2088	1907	1978	1978	2105	2061	2062	32,118
10-20	6566	6367	8164	11289	20113	41443	63472	39532	8343	8343	8403	7854	6301	5840	7597	6814	256,441
20-30	7553	3983	7885	15139	57681	106838	153687	141065	21582	12258	10978	5032	7822	9043	8848	8104	577,498
30-40	6255	5478	10556	12250	76975	179316	215331	84469	20553	9667	6766	5156	9113	12680	12372	8291	675,228
40-50	5785	6203	7667	17528	41018	279177	219807	38543	16740	9563	8208	7157	11549	16499	9075	4967	699,486
TOTAL	28,894	24,772	36,916	58,916	198,504	609,523	654,971	305,941	69,841	42,604	36,947	27,862	37,438	46,792	40,592	30,899	2,251,412

2.3 Meteorology

2.3.1 General

Travelers Research Corporation analyzed the meteorology of the plant site. Initial design criteria related to meteorology were based on data taken at St. Cloud and Minneapolis. Since the original Facility Description and Safety Analysis Report was written, a meteorological program was established to provide actual on-site meteorological data. The data obtained from this program are summarized in USAR Tables 2.3-5 through 2.3-21. These data confirm the adequacy of the initial design criteria used in the plant design.

The general climatic regime of the site is that of a marked continental type characterized by wide variations in temperature, scanty winter precipitation, normally ample summer rainfall, and a general tendency to extremes in all climatic features. Of special interest are the extremes in annual snowfall, which may be as little as six inches or as much as 88 inches; a temperature range of 145°F for the period of record; occasional severe thunderstorms with heavy rainfall and high winds; and the possibility of an occasional tornado or ice storm. These and other pertinent meteorological data are presented in the following sections.

2.3.2 Temperature

Average and extreme monthly air temperatures for the Monticello site are not available, but 54 years of data for St. Cloud and Minneapolis - St. Paul have been adjusted to give representative average values for the site area. The site is approximately 13 miles closer to St. Cloud than to Minneapolis. A summary of monthly air temperatures from January to December is given in Table 2.3-1.

2.3.3 Precipitation

Precipitation in the Monticello area is typical for the marked continental climate, with scanty winter precipitation and normally ample summer rainfall. The months of May through September have the greatest amounts of precipitation; average fall of rain during this period is 17-18 inches, or more than 70% of the annual rainfall. Thunderstorms are the principal source of rain during May through September and the Monticello area normally experiences 36 of these annually. The heaviest rainfall also occurs during a particularly severe thunderstorm. A summary of precipitation statistics is shown in Table 2.3-2 (based on St. Cloud and Minneapolis - St. Paul averages). Average monthly snowfall statistics are given in Table 2.3-3.

Intense rainfall is produced by an occasional severe thunderstorm. The return period of extreme short interval rainfall is a useful guide. The nearest location for which return period data are available and which should be reasonably representative for the Monticello area is Minneapolis. This data is shown in Figure 2.3-1.

Snow load data available from a Housing and Home Finance Agency (HHFA) study conducted in 1952 (Reference 18) are given in Table 2.3-4.

Data relating to freezing rain and resultant formation of glaze ice on highways and utility lines are available from the following studies:

American Telephone and Telegraph Company, 1917-18 to 1924-25 (Reference 19)
Edison Electric Institute, 1926-27 to 1937-38 (Reference 20)
Association of American Railroads, 1928-29 to 1936-37 (Reference 21)
Quartermaster Research and Engineering Command, U.S. Army, 1959 (Reference 22)

The U.S. Weather Bureau also maintains annual summaries. The following is a fairly accurate description of the glaze-ice climatology of middle Minnesota.

Time of occurrence - October through April
Average frequency without regard to ice thickness, 1-2 storms per year
Duration of ice on utility lines - 36 hours (mean) to 83 hours (maximum of record)

Return periods for freezing rain storms producing ice of various thickness are:

0.25 inch - Once every 2 years
0.50 inch - Once every 2 years
0.75 inch - Once every 3 years

2.3.4 Winds and Wind Loading

The preoperational meteorological data program is described in Sections 2.3.4 and 2.3.5 of the FSAR. The Monticello plant is currently provided with a 100-meter meteorological tower. Wind speed direction, and temperature difference instrumentation is located at approximately ten meters and at the elevation of the plant effluent point (43 meters and 100 meters). In addition, temperature, humidity, and rainfall instruments are provided. Meteorological data is used to compute dispersion (X/Q) and deposition (D/Q) factors for use in the dose assessment of airborne releases. Wind speed, direction, and atmosphere stability class are averaged over the release period and serve as inputs to a dispersion model. Stability class is determined using temperature difference measurements between the ten meter elevation and the elevation of the release.

Wind frequency distributions for the 10 and 100 meter tower elevations for the period January 1, 1980 through December 31, 1980 are presented in Tables

2.3-5 through 2.3-20. The distributions are for Stability A through G, as defined in Table 1 of the proposed revision 1 to Regulatory Guide 1.23 issued September 1980 (Reference 39). Annual average dispersion factor (X/Q) and deposition per unit area (D/Q) were computed for this period and are presented in Tables 2.3-22 through 2.3-27. NRC computer code XOQDOQ was used for these calculations (Reference 14). This historical data may be useful in estimating offsite doses due to routine releases of airborne radioactive effluents from the reactor building vent and plant stack.

2.3.4.1 Tornadoes and Severe Thunderstorms

Severe storms such as tornadoes are not numerous, but they do occur occasionally. The latitude of the Monticello site places it at the northern edge of the region of maximum tornado frequency in the United States, but only a few tornadoes have occurred in this vicinity. Eight tornadoes have been reported in Wright County during the period 1916-1967, two of which subsequently moved across the Mississippi River into Sherburne County.

A 1-degree square[1], lying between 45 and 46 degrees north, and between 93 and 94 degrees west, encompasses the Monticello site. There have been approximately eight tornado occurrences reported in this 1-degree square in the 14-year test period, 1953-1966. The ratio of eight tornadoes in 14 years gives a mean annual tornado frequency of 0.6. This frequency is confirmed by the Mean Annual Tornado Frequency figures published by the U.S. Department of Commerce, Weather Bureau (Reference 31).

Using the methods described by H. C. S. Thom (Reference 2), with a mean annual tornado frequency of 0.6, the probability of a tornado striking a given point in the outlined 1-degree square, which encompasses the Monticello site, can be calculated to be 5×10^{-4} per year, or one tornado every 2000 years. The effects of the tornado phenomenon including possible effects of missiles and water loss effects in the fuel pool are discussed in Reference 3 of this section.

Subsequently, it was determined the drywell head could become a missile hazard for the spent fuel pool, however, since the probability is less than 10^{-7}, it is not a credible missile.

The average number of thunderstorms for Minneapolis and St. Cloud is 36 with more than half of these occurring in June, July, and August. Therefore, it is expected that the Monticello site may experience an average of 36 thunderstorms annually. The fastest wind recorded for 54 years of record for each month at Minneapolis is given in Table 2.3-21.

2.3.4.2 Conclusions

The meteorology of the site area is basically that of a marked continental area with relatively favorable atmospheric dilution conditions prevailing. Diffusion

1. In this area, a 1-degree square is approximately 3,354 square miles.

climatology comparisons with other locations indicate that the site is typical of the North Central United States. Frequency of inversion is expected to be 30-40% of the year.

The site is located in an area occasionally traversed by storms and tornadoes. Maximum reported wind speed associated with passage of storm is 92 mph.

2.3.5 Plant Design Based on Meteorology

The station is designed with an off-gas stack to be used for continuous dispersal of gases to the atmosphere. Based on meteorological data at the site, plant operational characteristics, and stack design, the off-site doses arising from routine plant operation will satisfy the guidelines of Appendix I to 10CFR50.

A listing of other relevant reference material is given in References 4 through 9.

Class I and Class II Station structures are designed to withstand the effects of 100 mph winds at 30-feet above ground with a gust factor of 1.1. Structures and systems which are necessary for a safe shutdown of the reactor and maintaining a shutdown condition are designed to withstand tornado wind loadings of 300 mph.

Bibliography:

Rainfall Intensity - Duration - Frequency Curves, Tech. Paper No. 25, U.S. Weather Bureau (1955) (Reference 23).

Climatological Data with Comparative Data, Minneapolis - St. Paul, Minnesota, 1953-1956 - U.S. Weather Bureau (2 publications) (Reference 24).

Climatological Data with Comparative Data, St. Cloud, Minnesota 1953-1965 - U.S. Weather Bureau (2 publications) (Reference 25).

Climatography of the United States, No. 86-17, Minnesota, U.S. Weather Bureau (Reference 26).

Local Climatological Data with Comparative Data, 1965 - U.S. Weather Bureau (Reference 27).

"Snow Load Studies", Housing Research Paper 19, Housing and Home Finance Agency, 1952 (Reference 28).

"Glaze, Its Meteorology and Climatology, Geographical Distribution and Economic Effects, " Quartermaster Research and Engineering Center, 1959 (Reference 29).

Climatography of the United States No. 60-21, Minnesota - U.S. Weather Bureau (Reference 30).

Table 2.3-1 Monthly Air Temperature

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
Maximum	21	24	38	55	68	77	83	80	72	59	40	26
Minimum	3	6	20	35	46	56	61	59	50	39	24	10
Mean	12	15	29	45	57	66	72	70	61	49	32	18
Extreme Maximum	59	61	82	91	105	103	107	104	105	90	75	63
Extreme Minimum	-38	-34	-30	4	20	33	42	38	22	8	-18	-29

Table 2.3-2 Summary of Precipitation Statistics

Month	Days with 0.01 inch or more	Mean (inches)	Extreme Monthly Min. (inches)	Extreme Monthly Max. (inches)	*Max. in 24 hours (inches)	Days with Thunder-storms
Dec	9	0.77	T	2.48	1.05	0
Jan	8	0.78	0.02	2.82	1.90	0
Feb	7	0.80	0.01	3.10	1.83	0
Winter	24	2.35	-	-	-	0
March	10	1.32	0.11	3.95	2.00	1
April	9	1.94	0.32	5.72	3.15	2
May	12	3.11	0.20	10.00	5.00	5
Spring	31	6.37	-	-	-	8
June	13	4.06	0.87	9.78	3.35	8
July	10	2.86	0.31	12.34	4.80	7
Aug	10	2.83	0.31	8.99	4.62	6
Summer	33	9.75	-	-	-	21
Sept	9	2.92	0.24	9.24	3.65	4
Oct	8	1.65	.01	7.18	3.24	2
Nov	8	1.40	.01	4.66	1.44	1
Fall	25	5.97	T	-	-	7
Annual	113	24.44				

* St. Cloud 1894-1965
T = TRACE

I/djm

Table 2.3-3 Average Monthly Snowfall (inches)

Location	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Ann
Minneapolis St. Paul	6.3	8.0	11.5	2.7	0.2	0.0	0.0	0.0	0.1	0.3	6.1	7.0	42.2
St. Cloud	6.5	7.7	11.5	2.8	0.1	0.0	0.0	0.0	0.1	0.4	6.3	7.0	42.4

Maximum in 24 hours: Minneapolis 16.2 inches
St. Cloud 12.2 inches

Table 2.3-4 Snow Load Data

Location	Wt. of Seasonal Snowpack Equalled or Exceeded 1 Yr in 10	Wt. of Max Snowpack of Record	Wt. of Estimated Max. Accumulation on Grd plus Wt. of Max. Possible Storm
Minneapolis	30 lb/ft^2	40 lb/ft^2	50 lb/ft^2
St. Cloud	30 lb/ft^2	40 lb/ft^2	50 lb/ft^2

Table 2.3-5 Wind Frequency Distributions at 10 Meter Level, Stability Class A
(Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	3	20	34	15	4	0	76
NNE	4	11	11	2	0	0	28
NE	5	17	23	1	0	0	46
ENE	9	25	13	0	0	0	47
E	4	18	12	3	1	0	38
ESE	4	24	32	7	1	0	68
SE	4	22	43	24	0	0	93
SSE	3	13	47	32	7	0	102
S	2	18	39	36	26	0	121
SSW	3	25	60	26	3	0	117
SW	2	21	43	10	0	0	76
WSW	5	27	34	18	1	0	85
W	3	25	12	15	4	0	59
WNW	5	21	34	22	5	0	87
NW	4	20	51	27	7	0	109
NNW	2	10	37	30	5	0	84
VAR	0	0	0	0	0	0	0

Total Hours this Class	1242
Hours of Calm this Class	6
Percent of all Data this Class	15.14

Table 2.3-6 Wind Frequency Distributions at 10 Meter Level, Stability Class B (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	1	7	11	3	0	0	22
NNE	0	6	4	0	1	0	11
NE	1	4	5	1	0	0	11
ENE	0	5	0	0	0	0	5
E	0	4	0	0	0	0	4
ESE	0	4	4	1	1	0	10
SE	0	4	2	1	1	0	8
SSE	1	5	3	3	2	0	14
S	3	5	3	3	0	0	14
SSW	2	2	7	2	0	0	13
SW	4	2	4	0	0	0	10
WSW	1	5	5	1	0	0	12
W	0	1	4	2	0	0	7
WNW	1	7	8	2	1	0	19
NW	1	7	9	6	3	0	26
NNW	1	8	8	4	1	0	22
VAR	0	0	0	0	0	0	0

Total Hours this Class	208
Hours of Calm this Class	0
Percent of all Data this Class	2.54

Table 2.3-7 Wind Frequency Distributions at 10 Meter Level, Stability Class C (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	4	14	15	5	2	0	40
NNE	1	7	11	1	0	0	20
NE	2	7	5	1	0	0	15
ENE	0	11	0	0	0	0	11
E	1	5	1	0	0	0	7
ESE	2	6	6	1	1	0	16
SE	0	5	8	2	2	0	17
SSE	0	7	6	7	0	0	20
S	1	5	9	4	1	1	21
SSW	0	6	4	1	0	1	12
SW	2	8	11	4	0	0	25
WSW	0	8	6	0	1	0	15
W	0	7	3	3	2	0	15
WNW	2	4	14	7	1	0	28
NW	2	1	12	2	1	0	18
NNW	0	8	16	8	0	0	32
VAR	0	0	0	0	0	0	0

Total Hours this Class	313
Hours of Calm this Class	1
Percent of all Data this Class	3.82

Table 2.3-8 Wind Frequency Distributions at 10 Meter Level, Stability Class D (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	37	83	118	62	10	0	310
NNE	19	56	55	18	3	0	151
NE	26	56	61	12	0	0	155
ENE	24	71	28	1	0	0	124
E	12	58	47	9	0	0	126
ESE	13	75	79	34	0	0	201
SE	11	63	123	40	6	0	243
SSE	13	35	80	14	1	0	143
S	11	34	53	26	6	0	130
SSW	8	31	36	8	4	1	88
SW	5	23	27	3	2	0	60
WSW	9	18	24	4	3	0	58
W	7	28	20	15	3	0	78
WNW	5	40	72	29	20	3	169
NW	17	37	95	55	25	1	230
NNW	26	69	170	108	14	0	387
VAR	0	0	0	0	0	0	0

Total Hours this Class	2753
Hours of Calm this Class	100
Percent of all Data this Class	33.56

Table 2.3-9 Wind Frequency Distributions at 10 Meter Level, Stability Class E
(Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	28	96	48	7	0	0	179
NNE	15	39	17	2	0	0	73
NE	19	50	21	3	0	0	93
ENE	17	30	13	1	0	0	61
E	14	35	19	1	0	0	69
ESE	13	61	45	2	0	0	121
SE	12	70	49	3	0	0	134
SSE	9	50	38	15	1	0	113
S	10	32	33	28	2	0	105
SSW	13	35	41	22	1	0	112
SW	15	21	18	5	0	0	59
WSW	15	28	14	11	0	0	68
W	18	43	30	2	0	0	93
WNW	9	101	98	22	0	0	230
NW	11	54	87	36	2	0	190
NNW	20	87	113	33	4	0	257
VAR	0	0	0	0	0	0	0

Total Hours this Class	2008
Hours of Calm this Class	51
Percent of all Data this Class	24.48

Table 2.3-10 Wind Frequency Distributions at 10 Meter Level, Stability Class F (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	29	35	2	0	0	0	66
NNE	8	14	2	0	0	0	24
NE	18	14	2	0	0	0	34
ENE	14	9	0	0	0	0	23
E	12	26	0	0	0	0	38
ESE	14	46	6	0	0	0	66
SE	9	40	6	5	0	0	60
SSE	15	36	9	2	2	1	65
S	9	29	19	0	0	0	57
SSW	14	33	8	2	0	0	57
SW	20	25	6	0	0	0	51
WSW	18	39	3	1	0	0	61
W	18	37	7	0	0	0	62
WNW	15	31	0	0	0	0	46
NW	17	29	10	0	0	0	56
NNW	14	69	11	0	0	0	94
VAR	0	0	0	0	0	0	0

Total Hours this Class	871
Hours of Calm this Class	11
Percent of all Data this Class	10.62

Table 2.3-11 Wind Frequency Distributions at 10 Meter Level, Stability Class G (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	43	23	1	0	0	0	67
NNE	16	7	1	0	0	0	24
NE	17	12	0	0	0	0	29
ENE	15	1	0	0	0	0	16
E	15	5	0	0	0	0	20
ESE	17	10	0	0	0	0	27
SE	18	14	0	0	0	0	32
SSE	35	30	0	0	0	0	65
S	33	44	6	0	0	0	83
SSW	49	35	3	0	0	0	87
SW	35	14	0	0	0	0	49
WSW	38	28	0	0	0	0	66
W	33	22	0	0	0	0	55
WNW	32	11	0	0	0	0	43
NW	26	19	0	0	0	0	45
NNW	41	30	0	0	0	0	71
VAR	0	0	0	0	0	0	0

Total Hours this Class	808
Hours of Calm this Class	29
Percent of all Data this Class	9.85

Table 2.3-12 Wind Frequency Distributions at 10 Meter Level, All Classes Combined (Hours at Each Wind Speed and Direction)

(Page 1 of 2)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	145	278	229	92	16	0	760
NNE	63	140	101	23	4	0	331
NE	88	160	117	18	0	0	383
ENE	79	152	54	2	0	0	287
E	58	151	79	13	1	0	302
ESE	63	226	172	45	3	0	509
SE	54	218	231	75	9	0	587
SSE	76	176	183	73	13	1	522
S	69	167	162	97	35	1	531
SSW	89	167	159	61	8	2	486
SW	83	114	109	22	2	0	330
WSW	86	153	86	35	5	0	365
W	79	163	76	37	14	0	369
WNW	69	215	226	82	27	3	622
NW	78	167	264	126	38	1	674
NNW	104	281	355	183	24	0	947
VAR	0	0	0	0	0	0	0

Data Recovery Summary for Period

Total Hours	8784
Hours of Calm	198
Hours of Bad Data	581
Percent Data Recovery	93.39

l/djm

Table 2.3-12 Wind Frequency Distributions at 10 Meter Level, All Classes Combined (Hours at Each Wind Speed and Direction)

(Page 2 of 2)

Percent Acceptable Observations in Each Stability Class

Class A	15.14
Class B	2.54
Class C	3.82
Class D	33.56
Class E	24.48
Class F	10.62
Class G	9.85

Average Wind Speed for Each Wind Category

1 to 3 MPH	2.4
4 to 7 MPH	5.5
8 to 12 MPH	9.7
13 to 18 MPH	14.7
19 to 24 MPH	20.5
Above 24 MPH	25.8

Table 2.3-13 Wind Frequency Distributions at 100 Meter Level, Stability Class A
(Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	1	3	9	7	6	7	33
NNE	2	3	0	0	0	0	5
NE	0	1	0	1	0	0	2
ENE	0	2	2	0	0	0	4
E	0	0	0	1	0	0	1
ESE	0	6	7	16	3	2	34
SE	0	7	8	24	13	4	56
SSE	0	1	10	32	21	1	65
S	0	3	10	28	18	7	66
SSW	0	3	16	23	16	8	66
SW	1	6	9	16	6	2	40
WSW	0	1	9	24	18	0	52
W	0	3	8	8	17	3	39
WNW	1	1	4	2	7	4	19
NW	1	2	4	11	7	1	26
NNW	0	1	5	17	9	1	33
VAR	0	0	0	0	0	0	0

Total Hours this Class	656
Hours of Calm this Class	115
Percent of all Data this Class	7.98

Table 2.3-14 Wind Frequency Distributions at 100 Meter Level, Stability Class B (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	0	4	15	16	4	0	39
NNE	0	4	5	10	0	0	19
NE	0	3	10	3	0	0	16
ENE	1	3	6	1	0	0	11
E	0	2	3	0	0	0	5
ESE	0	3	7	2	2	1	15
SE	0	2	8	3	3	0	16
SSE	0	1	14	9	2	1	27
S	0	5	8	5	4	1	23
SSW	1	2	14	9	7	1	34
SW	1	4	14	5	2	0	26
WSW	0	4	6	5	5	0	20
W	0	5	6	4	4	3	22
WNW	0	2	4	2	1	5	14
NW	0	3	7	8	11	1	30
NNW	0	4	11	8	9	0	32
VAR	0	0	0	0	0	0	0

Total Hours this Class	349
Hours of Calm this Class	0
Percent of all Data this Class	4.25

Table 2.3-15 Wind Frequency Distributions at 100 Meter Level, Stability Class C (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	0	3	16	15	2	1	37
NNE	0	5	13	2	0	0	20
NE	0	2	2	4	0	0	8
ENE	0	4	11	0	0	0	15
E	0	3	9	2	1	0	15
ESE	0	8	6	5	0	0	19
SE	0	4	1	3	2	0	10
SSE	1	1	9	5	3	0	19
S	0	3	7	1	2	2	15
SSW	0	6	13	7	4	1	31
SW	0	4	4	6	1	1	16
WSW	0	4	7	7	0	0	18
W	0	4	4	5	3	1	17
WNW	2	3	11	7	5	7	35
NW	1	3	12	21	4	4	45
NNW	3	11	10	10	4	3	41
VAR	0	0	0	0	0	0	0

Total Hours this Class	361
Hours of Calm this Class	0
Percent of all Data this Class	4.39

Table 2.3-16 Wind Frequency Distributions at 100 Meter Level, Stability Class D (Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	17	46	84	120	101	49	417
NNE	15	38	45	67	19	3	187
NE	10	21	36	37	18	6	128
ENE	6	36	60	34	4	1	141
E	10	45	51	25	12	5	148
ESE	12	39	59	56	44	15	225
SE	9	27	51	130	69	20	306
SSE	4	30	51	76	26	14	201
S	7	15	50	60	18	11	161
SSW	11	25	40	39	32	7	154
SW	6	22	25	28	16	8	105
WSW	6	17	17	33	9	7	89
W	5	27	15	22	18	15	102
WNW	13	26	47	61	48	41	236
NW	8	23	52	100	95	63	341
NNW	10	45	90	151	120	82	498
VAR	0	0	0	0	0	0	0

Total Hours this Class	3504
Hours of Calm this Class	65
Percent of all Data this Class	42.64

Table 2.3-17 Wind Frequency Distributions at 100 Meter Level, Stability Class E
(Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	2	16	36	80	54	12	200
NNE	1	12	20	51	17	1	102
NE	2	12	20	29	12	3	78
ENE	0	12	42	19	7	1	81
E	5	7	35	30	7	0	84
ESE	4	10	21	39	20	3	97
SE	0	8	25	61	32	4	130
SSE	2	9	27	76	40	5	159
S	2	14	30	36	36	18	136
SSW	1	4	23	43	52	20	143
SW	2	8	10	20	53	7	100
WSW	3	18	17	20	22	2	82
W	2	13	21	29	18	3	86
WNW	2	6	31	66	55	4	164
NW	2	14	29	75	50	2	172
NNW	3	15	31	68	67	11	195
VAR	0	0	0	0	0	0	0

Total Hours this Class	2032
Hours of Calm this Class	23
Percent of all Data this Class	24.73

Table 2.3-18 Wind Frequency Distributions at 100 Meter Level, Stability Class F
(Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	3	9	14	27	18	2	73
NNE	0	5	17	16	13	0	51
NE	1	6	22	13	7	1	50
ENE	0	6	21	14	0	0	41
E	2	6	13	18	3	0	42
ESE	0	6	9	18	7	1	41
SE	2	8	12	22	18	0	62
SSE	2	5	13	30	21	3	74
S	2	8	8	30	12	7	67
SSW	0	2	9	21	33	2	67
SW	1	2	8	42	30	0	83
WSW	2	8	10	19	23	5	67
W	1	6	17	14	10	1	49
WNW	3	8	17	37	11	1	77
NW	4	10	22	33	5	0	74
NNW	5	14	22	37	4	0	82
VAR	0	0	0	0	0	0	0

Total Hours this Class	1000
Hours of Calm this Class	0
Percent of all Data this Class	12.17

Table 2.3-19 Wind Frequency Distributions at 100 Meter Level, Stability Class G
(Hours at Each Wind Speed and Direction)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	0	7	8	5	2	0	22
NNE	0	2	14	7	2	0	25
NE	1	3	7	6	2	0	19
ENE	1	3	9	1	0	0	14
E	0	2	5	6	0	0	13
ESE	0	3	3	5	0	0	11
SE	0	0	8	8	3	0	19
SSE	3	5	2	5	2	0	17
S	0	2	3	2	0	0	7
SSW	0	2	5	11	1	0	19
SW	0	8	13	7	7	0	35
WSW	3	4	11	3	4	1	26
W	0	3	13	6	2	0	24
WNW	0	3	11	5	4	0	23
NW	2	6	8	9	0	0	25
NNW	1	5	5	2	2	2	17
VAR	0	0	0	0	0	0	0

Total Hours this Class	316
Hours of Calm this Class	0
Percent of all Data this Class	3.85

Table 2.3-20 Wind Frequency Distributions at 100 Meter Level, All Classes Combined (Hours at Each Wind Speed and Direction)

(Page 1 of 2)

Wind Speed (MPH)

WIND DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
N	23	88	182	270	187	71	821
NNE	18	69	114	153	51	4	409
NE	14	48	97	93	39	10	301
ENE	8	66	151	69	11	2	307
E	17	65	116	82	23	5	308
ESE	16	75	112	141	76	22	442
SE	11	56	113	251	140	28	599
SSE	12	52	126	233	115	24	562
S	11	50	116	162	90	46	475
SSW	13	44	120	153	145	39	514
SW	11	54	83	124	115	18	405
WSW	14	56	77	111	81	15	354
W	8	61	84	88	72	26	339
WNW	21	49	125	180	131	62	568
NW	18	61	134	257	172	71	713
NNW	22	95	174	293	215	99	898
VAR	0	0	0	0	0	0	0

Data Recovery Summary for Period

Total Hours	8784
Hours of Calm	203
Hours of Bad Data	566
Percent Data Recovery	93.56

l/djm

Table 2.3-20 Wind Frequency Distributions at 100 Meter Level, All Classes Combined (Hours at Each Wind Speed and Direction)

(Page 2 of 2)

Percent Acceptable Observations in Each Stability Class

Class A	7.98
Class B	4.25
Class C	4.39
Class D	42.64
Class E	24.73
Class F	12.17
Class G	3.85

Average Wind Speed for Each Wind Category

1 to 3 MPH	2.6
4 to 7 MPH	5.7
8 to 12 MPH	10.2
13 to 18 MPH	15.5
19 to 24 MPH	21.1
Above 24 MPH	28.2

Table 2.3-21 Maximum Wind Velocity

Month	Speed, MPH	Direction	Year
Jan	47	NW	1928
Feb	52	NW	1952
March	56	SW	1920
April	58	N	1912
May	61	NW	1964
June	63	NW	1939
July	92*	W	1951
August	57	NW	1922
September	50	NW	1921
October	73	S	1949
November	60	SW	1959
December	52	W	1946

* Associated with the July 20, 1951 tornado

l/djm

Table 2.3-22 Annual Average Dispersion Factor (X/Q) - Reactor Building Vent Releases

Reactor Building Vent
No Decay, Undepleted
Corrected for Open Terrain Recirculation

Annual Average CHI/Q (Sec/Meter Cubed) — Distance in Miles

Sector	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	6.345E-06	2.532E-06	1.812E-06	1.206E-06	7.098E-07	4.539E-07	3.211E-07	2.736E-07	2.433E-07	2.106E-07	1.864E-07
SSW	2.742E-06	1.163E-06	8.628E-07	5.724E-07	3.233E-07	2.500E-07	2.108E-07	1.543E-07	1.192E-07	1.011E-07	8.773E-08
SW	2.985E-06	1.246E-06	9.472E-07	6.498E-07	3.851E-07	3.108E-07	2.672E-07	2.090E-07	1.704E-07	1.497E-07	1.320E-07
WSW	1.949E-06	8.250E-07	6.662E-07	4.821E-07	3.037E-07	2.462E-07	2.106E-07	1.548E-07	1.198E-07	1.071E-07	9.643E-08
W	2.393E-06	9.695E-07	7.325E-07	5.018E-07	3.014E-07	2.422E-07	2.084E-07	1.631E-07	1.329E-07	1.061E-07	8.733E-08
WNW	4.552E-06	1.768E-06	1.247E-06	8.060E-07	4.532E-07	3.477E-07	2.900E-07	2.393E-07	2.020E-07	1.594E-07	1.300E-07
NW	5.502E-06	2.094E-06	1.399E-06	8.565E-07	4.435E-07	2.855E-07	2.046E-07	1.688E-07	1.459E-07	1.235E-07	1.071E-07
NNW	4.704E-06	1.698E-06	1.112E-06	6.930E-07	3.859E-07	2.493E-07	1.796E-07	1.386E-07	1.121E-07	9.375E-08	8.041E-08
N	5.225E-06	1.822E-06	1.133E-06	6.806E-07	3.661E-07	2.315E-07	1.643E-07	1.347E-07	1.163E-07	9.604E-08	8.136E-08
NNE	4.357E-06	1.489E-06	9.479E-07	5.946E-07	3.437E-07	2.255E-07	1.642E-07	1.275E-07	1.035E-07	8.665E-07	7.431E-08
NE	2.523E-06	9.147E-07	5.967E-07	3.771E-07	2.148E-07	1.592E-07	1.290E-07	1.011E-07	8.234E-08	6.909E-08	5.929E-08
ENE	3.074E-06	1.035E-06	6.587E-07	4.245E-07	2.560E-07	1.829E-07	1.424E-07	1.119E-07	9.141E-08	7.688E-08	6.611E-08
E	3.142E-06	1.104E-06	7.441E-07	4.922E-07	2.963E-07	1.999E-07	1.471E-07	1.146E-07	9.290E-08	7.763E-08	6.638E-08
ESE	5.744E-06	2.195E-06	1.425E-06	8.550E-07	4.320E-07	2.693E-07	1.880E-07	1.411E-07	1.112E-07	9.091E-08	7.636E-08
SE	6.575E-06	2.438E-06	1.529E-06	8.966E-07	4.458E-07	2.949E-07	2.192E-07	1.638E-07	1.287E-07	1.049E-07	8.790E-08
SSE	9.467E-06	3.635E-06	2.343E-06	1.395E-06	7.007E-07	4.363E-07	3.045E-07	2.284E-07	1.801E-07	1.473E-07	1.239E-07

Annual Average CHI/Q (Sec/Meter Cubed) — Distance in Miles

Sector	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	1.584E-07	8.944E-08	6.152E-08	3.795E-08	2.685E-08	2.049E-08	1.641E-08	1.359E-08	1.155E-08	9.997E-09	8.787E-09
SSW	7.398E-08	4.073E-08	2.760E-08	1.673E-08	1.170E-08	8.858E-09	7.051E-09	5.812E-09	4.916E-09	4.241E-09	3.715E-09
SW	1.104E-07	5.913E-08	3.946E-08	2.349E-08	1.626E-08	1.223E-08	9.682E-09	7.949E-09	6.701E-09	5.765E-09	5.040E-09
WSW	8.102E-08	4.410E-08	2.971E-08	1.787E-08	1.244E-08	9.379E-09	7.442E-09	6.118E-09	5.163E-09	4.445E-09	3.888E-09
W	7.362E-08	4.039E-08	2.729E-08	1.647E-08	1.150E-08	8.698E-09	6.922E-09	5.706E-09	4.827E-09	4.165E-09	3.650E-09
WNW	1.087E-07	5.814E-08	3.870E-08	2.297E-08	1.588E-08	1.194E-08	9.459E-09	7.772E-09	6.557E-09	5.645E-09	4.939E-09
NW	9.039E-07	4.975E-08	3.367E-08	2.037E-08	1.424E-08	1.079E-08	8.595E-09	7.093E-09	6.006E-09	5.187E-09	4.550E-09
NNW	6.954E-08	4.177E-08	2.987E-08	1.936E-08	1.413E-08	1.103E-08	8.994E-09	7.559E-08	6.498E-09	5.684E-09	5.041E-09
N	7.033E-08	4.216E-08	3.010E-08	1.946E-08	1.419E-08	1.108E-08	9.028E-09	7.587E-09	6.523E-09	5.706E-09	5.061E-09
NNE	6.492E-08	4.041E-08	2.954E-08	1.967E-08	1.461E-08	1.155E-08	9.510E-09	8.057E-09	6.972E-09	6.134E-09	5.467E-09
NE	5.180E-08	3.212E-08	2.336E-08	1.544E-08	1.141E-08	8.987E-09	7.377E-09	6.234E-09	5.384E-09	4.728E-09	4.207E-09
ENE	5.786E-08	3.612E-08	2.639E-08	1.753E-08	1.298E-08	1.024E-08	8.412E-09	7.113E-09	6.146E-09	5.398E-09	4.805E-09
E	5.781E-08	3.546E-08	2.563E-08	1.681E-08	1.236E-08	9.700E-09	7.940E-09	6.694E-09	5.770E-09	5.058E-09	4.495E-09
ESE	6.554E-08	3.835E-08	2.701E-08	1.722E-08	1.248E-08	9.695E-09	7.881E-09	6.611E-09	5.675E-09	4.959E-09	4.394E-09
SE	7.530E-08	4.381E-08	3.074E-08	1.947E-08	1.401E-08	1.082E-08	8.747E-09	7.302E-09	6.241E-09	5.432E-09	4.797E-09
SSE	1.065E-07	6.296E-08	4.487E-08	2.923E-08	2.100E-08	1.621E-08	1.317E-08	1.105E-08	9.685E-09	8.630E-09	7.607E-09

I/djm

Table 2.3-23 Annual Average Dispersion Factor (X/Q) - Plant Stack Releases

Offgas Stack
No Decay, Undepleted
Corrected for Open Terrain Recirculation

Annual Average CHI/Q (Sec/Meter Cubed) — Distance in Miles

Sector	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	2.115E-07	4.610E-07	2.388E-07	1.593E-07	1.288E-07	9.864E-08	7.790E-08	6.894E-08	6.157E-08	5.380E-08	4.765E-08
SSW	2.837E-07	7.831E-07	3.300E-07	1.700E-07	1.106E-07	9.136E-08	7.844E-08	6.159E-08	5.017E-08	4.336E-08	3.810E-08
SW	1.845E-08	3.655E-08	3.938E-08	3.921E-08	3.866E-08	4.136E-08	4.103E-08	3.536E-08	3.093E-08	2.878E-08	2.690E-08
WSW	2.433E-08	4.174E-08	4.948E-08	4.936E-08	4.708E-08	4.665E-08	4.328E-08	3.408E-08	2.772E-08	2.516E-08	2.304E-08
W	5.617E-09	2.206E-08	3.707E-08	4.484E-08	5.007E-08	5.511E-08	5.516E-08	4.752E-08	4.142E-08	3.487E-08	2.990E-08
WNW	1.006E-07	6.505E-08	6.450E-08	6.468E-08	6.394E-08	6.555E-08	6.264E-08	5.602E-08	5.023E-08	4.155E-08	3.514E-08
NW	1.418E-07	6.927E-08	5.869E-08	5.975E-08	5.870E-08	5.118E-08	4.319E-08	3.917E-08	3.548E-08	3.103E-08	2.750E-08
NNW	1.477E-07	8.592E-08	6.979E-08	6.209E-08	5.752E-08	4.724E-08	3.884E-08	3.244E-08	2.757E-08	2.381E-08	2.085E-08
N	1.476E-07	8.231E-08	6.138E-08	5.204E-08	4.793E-08	3.936E-08	3.252E-08	2.897E-08	2.597E-08	2.233E-08	1.949E-08
NNE	1.582E-07	1.080E-07	8.621E-08	6.771E-08	5.532E-08	4.327E-08	3.479E-08	2.873E-08	2.427E-08	2.089E-08	1.825E-08
NE	2.384E-07	4.483E-07	1.951E-07	9.784E-08	5.879E-08	4.452E-08	3.628E-08	2.946E-08	2.468E-08	2.114E-08	1.844E-08
ENE	1.202E-07	7.218E-08	5.321E-08	3.986E-08	3.219E-08	2.775E-08	2.422E-08	2.069E-08	1.795E-08	1.577E-08	1.402E-08
E	9.542E-08	6.545E-08	5.063E-08	3.953E-08	3.280E-08	2.701E-08	2.253E-08	1.910E-08	1.645E-08	1.437E-08	1.271E-08
ESE	1.608E-07	4.092E-07	1.913E-07	1.103E-07	7.750E-08	5.977E-08	4.803E-08	3.978E-08	3.375E-08	2.917E-08	2.560E-08
SE	1.908E-07	4.410E-07	2.167E-07	1.285E-07	8.914E-08	7.234E-08	6.044E-08	4.895E-08	4.075E-08	3.467E-08	3.003E-08
SSE	8.598E-08	9.415E-08	1.104E-07	1.062E-07	9.305E-08	7.625E-08	6.228E-08	5.167E-08	4.366E-08	3.751E-08	3.271E-08

Annual Average CHI/Q (Sec/Meter Cubed) — Distance in Miles

Sector	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	4.135E-08	2.465E-08	1.726E-08	1.073E-08	7.580E-09	5.763E-09	4.597E-09	3.794E-09	3.211E-09	2.771E-09	2.428E-09
SSW	3.303E-08	1.972E-08	1.388E-08	8.722E-09	6.223E-09	4.774E-09	3.840E-09	3.193E-09	2.721E-09	2.363E-09	2.082E-09
SW	2.325E-08	1.368E-08	9.507E-09	5.853E-09	4.106E-09	3.106E-09	2.467E-09	2.029E-09	1.712E-09	1.474E-09	1.288E-09
WSW	1.978E-08	1.140E-08	7.837E-09	4.777E-09	3.339E-09	2.521E-09	2.001E-09	1.645E-09	1.387E-09	1.194E-09	1.043E-09
W	2.603E-08	1.561E-08	1.093E-08	6.765E-09	4.750E-09	3.590E-09	2.848E-09	2.339E-09	1.970E-09	1.692E-09	1.477E-09
WNW	3.025E-08	1.750E-08	1.201E-08	7.268E-09	5.039E-09	3.775E-09	2.976E-09	2.431E-09	2.039E-09	1.746E-09	1.519E-09
NW	2.380E-08	1.405E-08	9.773E-09	6.016E-09	4.216E-09	3.185E-09	2.527E-09	2.076E-09	1.750E-09	1.504E-09	1.313E-09
NNW	1.836E-08	1.147E-08	8.248E-09	5.287E-09	3.804E-09	2.929E-09	2.359E-09	1.961E-09	1.669E-09	1.448E-09	1.274E-09
N	1.724E-08	1.091E-08	7.886E-09	5.067E-09	3.640E-09	2.795E-09	2.244E-09	1.859E-09	1.578E-09	1.365E-09	1.197E-09
NNE	1.616E-08	1.027E-08	7.432E-09	4.768E-09	3.419E-09	2.619E-09	2.099E-09	1.736E-09	1.471E-09	1.271E-09	1.114E-09
NE	1.633E-08	1.048E-08	7.707E-09	5.102E-09	3.756E-09	2.945E-09	2.407E-09	2.027E-09	1.745E-09	1.528E-09	1.357E-09
ENE	1.259E-08	8.389E-09	6.250E-09	4.166E-09	3.064E-09	2.394E-09	1.948E-09	1.633E-09	1.400E-09	1.221E-09	1.079E-09
E	1.136E-09	7.459E-09	5.508E-09	3.633E-09	2.653E-09	2.062E-09	1.671E-09	1.396E-09	1.193E-09	1.038E-09	9.153E-10
ESE	2.276E-08	1.471E-08	1.080E-08	7.091E-09	5.173E-09	4.020E-09	3.258E-09	2.723E-09	2.328E-09	2.026E-09	1.789E-09
SE	2.640E-08	1.648E-08	1.188E-08	7.676E-09	5.564E-09	4.310E-09	3.490E-09	2.915E-09	2.493-09	2.170E-09	1.917E-09
SSE	2.889E-08	1.823E-08	1.318E-08	8.505E-09	6.071E-09	4.643E-09	3.728E-09	3.091E-09	2.651E-09	2.316E-09	2.028E-09

I/djm

Table 2.3-24 Relative Deposition per Unit Area (D/Q) - Reactor Building Vent Releases

Reactor Building Vent
Corrected for Open Terrain Recirculation

Relative Deposition per Unit Area (M**-2) at Fixed Points by Downwind Sectors

Distance in Miles

Sector	0.25	0.50	0.75	1.00	1.50	2.00	2.50	3.00	3.50	4.00	4.50
S	8.092E-08	3.151E-08	1.761E-08	8.946E-09	3.746E-09	1.900E-09	1.135E-09	7.688E-10	5.686E-10	4.375E-10	3.536E-10
SSW	3,154E-08	1.295E-08	7.461E-09	3.869E-09	1.609E-09	8.979E-10	5.352E-10	3.563E-10	2.567E-10	1.965E-10	1.583E-10
SW	3.300E-08	1.377E-08	7.966E-09	4.147E-09	1.735E-09	9.762E-10	5.841E-10	3.907E-10	2.836E-10	2.443E-10	2.629E-10
WSW	2.055E-08	9.475E-09	5.706E-09	3.047E-09	1.281E-09	7.625E-10	4.563E-10	3.047E-10	2.200E-10	1.693E-10	1.459E-10
W	2.502E-08	1.056E-08	6.179E-09	3.225E-09	1.349E-09	7.579E-10	4.517E-10	3.013E-10	2.184E-10	1.685E-10	1.366E-10
WNW	5.235E-08	2.088E-08	1.177E-08	5.991E-09	2.437E-09	1.320E-09	7.849E-10	5.228E-10	4.494E-10	3.415E-10	2.717E-10
NW	6.974E-08	2.703E-08	1.504E-08	7.583E-09	2.914E-09	1.492E-09	9.284E-10	6.290E-10	4.606E-10	3.515E-10	2.816E-10
NNW	6.209E-08	2.360E-08	1.286E-08	6.399E-09	2.543E-09	1.281E-09	7.729E-10	5.142E-10	3.680E-10	2.787E-10	2.281E-10
N	7.209E-08	2.676E-08	1.434E-08	7.046E-09	2.712E-09	1.364E-09	8.121E-10	5.491E-10	4.003E-10	3.078E-10	2.480E-10
NNE	5.609E-08	2.149E-08	1.150E-08	5.643E-09	2.168E-09	1.092E-09	6.510E-10	4.314E-10	3.073E-10	2.310E-10	1.807E-10
NE	3.345E-08	1.350E-08	7.297E-09	3.601E-09	1.354E-09	6.904E-10	4.220E-10	2.798E-10	1.994E-10	1.498E-10	1.171E-10
ENE	3.671E-08	1.447E-08	7.753E-09	3.811E-09	1.429E-09	7.286E-10	4.441E-10	2.946E-10	2.098E-10	1.573E-10	1.227E-10
E	3.616E-08	1.380E-08	7.441E-09	3.674E-09	1.383E-09	7.040E-10	4.220E-10	2.802E-10	1.993E-10	1.490E-10	1.157E-10
ESE	7.702E-08	2.887E-08	1.555E-08	7.653E-09	2.863E-09	1.450E-09	8.654E-10	5.727E-10	4.064E-10	3.034E-10	2.352E-10
SE	9.530E-08	3.536E-08	1.903E-08	9.380E-09	3.520E-09	1.787E-09	1.108E-09	7.322E-10	5.211E-10	3.917E-10	3.070E-10
SSE	1.223E-07	4.534E-08	2.479E-08	1.237E-08	4.704E-09	2.399E-09	1.438E-09	9.546E-10	6.786E-10	5.068E-10	3.929E-10

Distance in Miles

Sector	5.00	7.50	10.00	15.00	20.00	25.00	30.00	35.00	40.00	45.00	50.00
S	2.971E-10	1.641E-10	1.127E-10	6.546E-11	4.158E-11	2.793E-11	1.993E-11	1.486E-11	1.148E-11	9.147E-12	7.448E-12
SSW	1.323E-10	7.175E-11	4.870E-11	2.806E-11	1.782E-11	1.201E-11	8.596E-12	6.434E-12	4.986E-12	3.982E-12	3.250E-12
SW	2.105E-10	9.662E-11	5.959E-11	3.100E-11	1.909E-11	1.291E-11	9.317E-12	7.044E-12	5.505E-12	4.427E-12	3.634E-12
WSW	1.213E-10	6.451E-11	4.329E-11	2.471E-11	1.570E-11	1.062E-11	7.637E-12	5.741E-12	4.466E-12	3.580E-12	2.930E-12
W	1.154E-10	6.493E-11	4.495E-11	2.636E-11	1.680E-11	1.129E-11	8.061E-12	6.013E-12	4.646E-12	3.701E-12	3.013E-12
WNW	2.243E-10	1.166E-10	7.775E-11	4.381E-11	2.752E-11	1.845E-11	1.315E-11	9.804E-12	7.573E-12	6.024E-12	4.898E-12
NW	2.345E-10	1.257E-10	8.501E-11	4.874E-11	3.086E-11	2.075E-11	1.483E-11	1.109E-11	8.579E-12	6.843E-12	5.578E-12
NNW	1.892E-10	9.973E-11	6.677E-11	3.794E-11	2.401E-11	1.623E-11	1.168E-11	8.812E-12	6.897E-12	5.559E-12	4.588E-12
N	2.073E-10	1.125E-10	7.670E-11	4.423E-11	2.805E-11	1.887E-11	1.349E-11	1.008E-11	7.817E-12	6.237E-12	5.088E-12
NNE	1.461E-10	6.935E-11	4.359E-11	2.340E-11	1.477E-11	1.024E-11	7.634E-12	5.990E-12	4.874E-12	4.079E-12	3.502E-12
NE	9.447E-11	4.440E-11	2.767E-11	1.482E-11	9.433E-12	6.640E-12	5.023E-12	3.996E-12	3.291E-12	2.782E-12	2.410E-12
ENE	9.867E-11	4.581E-11	2.835E-11	1.505E-11	9.545E-12	6.726E-12	5.108E-12	4.086E-12	3.391E-12	2.886E-12	2.519E-12
E	9.243E-11	4.165E-11	2.516E-11	1.293E-11	8.073E-12	5.669E-12	4.320E-12	3.483E-12	2.928E-12	2.518E-12	2.228E-12
ESE	1.878E-10	8.431E-11	5.083E-11	2.596E-11	1.690E-11	1.118E-11	8.386E-12	6.635E-12	5.466E-12	4.613E-12	3.999E-12
SE	2.489E-10	1.199E-10	7.608E-11	4.100E-11	2.565E-11	1.747E-11	1.273E-11	9.745E-12	7.737E-12	6.321E-12	5.291E-12
SSE	3.136E-10	1.405E-10	8.434E-11	5.267E-11	3.273E-11	2.225E-11	1.622E-11	1.342E-11	1.457E-11	1.291E-11	1.047E-11

Table 2.3-25 Relative Deposition per Unit Area (D/Q) - Plant Stack Releases

Offgas Stack
Corrected for Open Terrain Recirculation

Relative Deposition per Unit Area (M**-2) at Fixed Points by Downwind Sectors

Distance in Miles

Sector	0.25	0.50	0.75	1.00	1.50	2.00	2.50	3.00	3.50	4.00	4.50
S	8.898E-09	7.159E-09	5.968E-09	4.054E-09	1.953E-09	1.193E-09	8.045E-10	5.771E-10	4.319E-10	3.739E-10	3.038E-10
SSW	5.688E-09	4.432E-09	3.478E-09	2.236E-09	1.025E-09	6.119E-10	4.918E-10	3.371E-10	2.454E-10	1.925E-10	1.510E-10
SW	1.821E-09	1.550E-09	1.419E-09	1.038E-09	5.308E-10	3.322E-10	2.875E-10	2.048E-10	1.483E-10	1.123E-10	8.902E-11
WSW	2.098E-09	1.769E-09	1.596E-09	1.155E-09	5.895E-10	3.655E-10	3.111E-10	2.140E-10	1.561E-10	1.241E-10	9.717E-11
W	1.487E-09	1.348E-09	1.350E-09	1.050E-09	5.609E-10	3.570E-10	3.036E-10	2.255E-10	1.633E-10	1.236E-10	9.681E-11
WNW	4.723E-09	3.809E-09	3.189E-09	2.174E-09	1.051E-09	6.427E-10	5.445E-10	3.870E-10	2.798E-10	2.117E-10	1.658E-10
NW	5.707E-09	4.661E-09	3.991E-09	2.772E-09	1.361E-09	8.380E-10	5.676E-10	4.081E-10	3.058E-10	2.692E-10	2.172E-10
NNW	7.648E-09	5.852E-09	4.428E-09	2.743E-09	1.212E-09	7.115E-10	4.696E-10	3.330E-10	2.477E-10	1.909E-10	1.511E-10
N	7.157E-09	5.428E-09	4.032E-09	2.450E-09	1.060E-09	6.161E-10	4.043E-10	2.858E-10	2.122E-10	1.634E-10	1.294E-10
NNE	8.998E-09	6.737E-09	4.863E-09	2.863E-09	1.196E-09	6.828E-10	4.434E-10	3.115E-10	2.307E-10	1.774E-10	1.404E-10
NE	6.944E-09	5.171E-09	3.688E-09	2.141E-09	8.802E-10	4.980E-10	3.217E-10	2.254E-10	1.666E-10	1.280E-10	1.013E-10
ENE	6.176E-09	4.591E-09	3.263E-09	1.885E-09	7.710E-10	4.350E-10	2.805E-10	1.963E-10	1.451E-10	1.115E-10	8.822E-11
E	5.361E-09	4.032E-09	2.939E-09	1.749E-09	7.403E-10	4.253E-10	2.773E-10	1.952E-10	1.447E-10	1.113E-10	8.813E-11
ESE	6.035E-09	4.770E-09	3.848E-09	2.538E-09	1.192E-09	7.196E-10	4.824E-10	3.449E-10	2.577E-10	1.989E-10	1.575E-10
SE	8.324E-09	6.599E-09	5.355E-09	3.552E-09	1.676E-09	1.014E-09	6.806E-10	4.870E-10	3.640E-10	2.810E-10	2.225E-10
SSE	7.413E-09	6.241E-09	5.616E-09	4.058E-09	2.056E-09	1.282E-09	8.739E-10	6.305E-10	4.732E-10	3.660E-10	2.897E-10

Distance in Miles

Sector	5.00	7.50	10.00	15.00	20.00	25.00	30.00	35.00	40.00	45.00	50.00
S	2.446E-10	1.114E-10	6.564E-11	3.308E-11	2.074E-11	1.466E-11	1.116E-11	8.927E-12	7.412E-12	6.290E-12	5.463E-12
SSW	1.217E-10	5.567E-11	3.292E-11	1.671E-11	1.053E-11	7.563E-12	5.836E-12	4.727E-12	3.969E-12	3.402E-12	2.986E-12
SW	7.162E-11	3.229E-11	1.879E-11	9.304E-12	5.797E-12	4.085E-12	3.138E-12	2.557E-12	2.174E-12	1.890E-12	1.690E-12
WSW	7.819E-11	3.531E-11	2.059E-11	1.022E-11	6.374E-12	4.485E-12	3.431E-12	2.776E-12	2.342E-12	2.020E-12	1.790E-12
W	7.788E-11	3.515E-11	2.044E-11	1.009E-11	6.253E-12	4.377E-12	3.335E-12	2.694E-12	2.270E-12	1.959E-12	1.739E-12
WNW	1.335E-10	6.042E-11	3.541E-11	1.776E-11	1.113E-11	7.835E-12	5.961E-12	4.770E-12	3.971E-12	3.378E-12	2.942E-12
NW	1.748E-10	7.937E-11	4.658E-11	2.335E-11	1.460E-11	1.028E-11	7.816E-12	6.248E-12	5.191E-12	4.410E-12	3.837E-12
NNW	1.222E-10	5.853E-11	3.613E-11	1.925E-11	1.225E-11	8.921E-12	6.812E-12	5.396E-12	4.378E-12	3.623E-12	3.046E-12
N	1.047E-10	5.019E-11	3.102E-11	1.657E-11	1.056E-11	7.732E-12	5.931E-12	4.702E-12	3.821E-12	3.165E-12	2.663E-12
NNE	1.137E-10	5.462E-11	3.383E-11	1.817E-11	1.163E-11	8.582E-12	6.621E-12	5.270E-12	4.294E-12	3.563E-12	3.002E-12
NE	8.210E-11	3.948E-11	2.448E-11	1.318E-11	8.450E-12	6.265E-12	4.847E-12	3.865E-12	3.153E-12	2.619E-12	2.207E-12
ENE	7.148E-11	3.439E-11	2.133E-11	1.149E-11	7.373E-12	5.474E-12	4.239E-12	3.382E-12	2.760E-12	2.293E-12	1.933E-12
E	7.135E-11	3.425E-11	2.120E-11	1.136E-11	7.261E-12	5.343E-12	4.113E-12	3.269E-12	2.661E-12	2.207E-12	1.858E-12
ESE	1.273E-10	6.078E-11	3.740E-11	1.979E-11	1.252E-11	8.999E-12	6.821E-12	5.363E-12	4.333E-12	3.575E-12	3.000E-12
SE	1.797E-10	8.583E-11	5.280E-11	2.792E-11	1.765E-11	1.268E-11	9.605E-12	7.548E-12	6.096E-12	5.029E-12	4.219E-12
SSE	2.339E-10	1.114E-10	6.831E-11	3.586E-11	2.255E-11	1.598E-11	1.200E-11	9.368E-12	7.532E-12	6.193E-12	5.185E-12

Table 2.3-26 Site Boundary X/Q and D/Q - Reactor Building Vent Releases

Reactor Building Vent
Corrected for Open Terrain Recirculation
Specific Points of Interest

Release ID	Type of Location	Sector	Distance (Miles)	Distance (Meters)	X/Q (Sec/Cub Meter) No Decay Undepleted	X/Q (Sec/Cub Meter) 2.260 Day Decay Undepleted	X/Q (Sec/Cub Meter) 8.000 Day Decay Depleted	D/Q (Per Sq Meter)
R	Site Boundary	S	0.34	547.	4.04E-06	4.03E-06	3.79E-06	5.36E-08
R	Site Boundary	SSW	0.32	515.	1.92E-06	1.92E--06	1.813-06	2.31E-08
R	Site Boundary	SW	0.32	515.	2.05E-06	2.05E-06	1.93E-06	2.43E-08
R	Site Boundary	WSW	0.35	563.	1.17E-06	1.17E-06	1.11E-06	1.43E-08
R	Site Boundary	W	0.48	772.	9.97E-07	9.96E-07	9.31E-07	1.11E-08
R	Site Boundary	WNW	0.68	1094.	1.33E-06	1.33E-06	1.24E-06	1.36E-06
R	Site Boundary	NW	0.43	692.	2.49E-06	2.49E-06	2.32E-06	3.34E-08
R	Site Boundary	NNW	0.53	853.	1.57E-06	1.57E-06	1.45E-06	2.17E-08
R	Site Boundary	N	0.51	821.	1.76E-06	1.75E-06	1.62E-06	2.60E-08
R	Site Boundary	NNE	0.58	933.	1.23E-06	1.22E-06	1.13E-06	1.72E-08
R	Site Boundary	NE	0.65	1046.	6.74E-07	6.73E-07	6.26E-07	9.13E-09
R	Site Boundary	ENE	0.83	1336.	5.55E-07	5.53E-07	5.14E-07	6.05E-09
R	Site Boundary	E	0.59	950.	9.09E-07	9.08E-07	8.39E-07	1.08E-08
R	Site Boundary	ESE	0.59	950.	1.81E-06	1.80E-06	1.67E-06	2.25E-08
R	Site Boundary	SE	0.61	982.	1.91E-06	1.91E-06	1.75E-06	2.62E-08
R	Site Boundary	SSE	0.43	692.	4.38E-06	4.38E-06	4.06E-06	5.65E-08

Table 2.3-27 Site Boundary X/Q and D/Q -Plant Stack Releases

Offgas Stack
Corrected for Open Terrain Recirculation
Specific Points of Interest

Release ID	Type of Location	Sector	Distance (Miles)	Distance (Meters)	X/Q (Sec/Cub Meter) No Decay Undepleted	X/Q (Sec/Cub Meter) 2.260 Day Decay Undepleted	X/Q (Sec/Cub Meter) 8.000 Day Decay Depleted	D/Q (Per Sq Meter)
O	Site Boundary	SSW	0.31	499.	6.50E-07	6.44E-07	6.48E-07	5.48E-09
O	Site Boundary	SW	0.33	531.	2.96E-08	2.96E-08	2.96E-08	1.75E-09
O	Site Boundary	SW	0.33	531.	2.96E-08	2.96E-08	2.96E-08	1.75E-09
O	Site Boundary	WSW	0.38	612.	3.54E-08	3.54E-08	3.54E-08	1.94E-09
O	Site Boundary	W	0.56	901.	2.49E-08	2.49E-08	2.46E-08	1.33E-09
O	Site Boundary	NW	0.78	1255.	5.70E-08	5.69E-08	5.61E-08	3.83E-09
O	Site Boundary	NW	0.53	853.	5.93E-08	5.92E-08	5.86E-08	4.55E-09
O	Site Boundary	NNW	0.61	982.	7.02E-08	7.02E-08	6.92E-08	5.12E-09
O	Site Boundary	N	0.59	950.	6.60E-08	6.60E-08	6.51E-08	4.83E-09
O	Site Boundary	N	0.63	1014.	6.33E-08	6.32E-08	6.23E-08	4.60E-09
O	Site Boundary	NNE	0.65	1046.	8.84E-08	8.83E-08	8.68E-08	5.49E-09
O	Site Boundary	ENE	0.78	1255.	4.96E-08	4.96E-08	4.86E-08	3.05E-09
O	Site Boundary	E	0.50	805.	6.12E-08	6.11E-08	6.06E-08	4.03E-09
O	Site Boundary	ESE	0.50	805.	3.42E-07	3.37E-07	3.37E-07	4.77E-09
O	Site Boundary	SSE	0.51	821.	9.11E-08	9.10E-08	9.02E-08	6.20E-09
O	Site Boundary	S	0.36	579.	4.78E-07	4.74E-07	4.77E-07	8.24E-09

2.4 Hydrology

2.4.1 Surface Water

The Monticello sites lies about one-third of the river distance from Elk River, Minnesota to St. Cloud, Minnesota. Stream flow records of the Mississippi were kept at Elk River by the U.S. Geological Survey. The gauging station at Elk River was about 2500 feet downstream from the confluence of the Elk River (the only significant river entering the Mississippi River between the cities of Elk River and St. Cloud) and the Mississippi River. The Elk River Station has closed and the U.S. Geological Survey established a gauging station on the Mississippi River at St. Cloud in 1989.

In Table 2.4-1, the number of years of record, the average annual flow, the minimum recorded flow, the maximum recorded flow at each gauging station are tabulated. From this data, and with information on Elk River flows, the following flow statistics are estimated for the Mississippi River at the Monticello site:

Average Flow - 4600 ft^3/sec
Minimum Flow - 240 ft^3/sec
Maximum Flow - 51,000 ft^3/sec

The average velocity of flow at the site varies between 1.5 to 2.5 ft/sec for flows below 10,000 cfs.

Figure 2.4-1 is a flow duration curve for the Mississippi River at St. Cloud. From this curve, the flow at Monticello is expected to exceed 1100 ft^3/sec 90% of the time, and 300 ft^3/sec 99% of the time.

Based on past temperature records from the Whitney Steam Plant at St. Cloud (since retired and removed) the average river temperature for these summer months is 71°F.

Because of possible low stream flow conditions, and high natural river water temperatures, two cooling towers are included in the plant design in order to meet the standards of the Minnesota Pollution Control Agency. At times of extremely low flow, the plant operates on a closed cycle and the makeup requirement of about 54 ft^3/sec is withdrawn from the river. At times of substantial flow and high ambient river temperature conditions, the cooling tower may be employed to control the temperature of discharged water.

All existing cooling towers are operated whenever the ambient river temperature measured at some point unaffected by the plant's discharge is consistently at or above 20°C (68°F), except in the event the cooling towers are out of service due to equipment failure or performance of maintenance to prevent equipment failure.

The spring flood of 1965 exceeds all flood flows on record to date. Figure 2.4-2 shows the location of three flood stage boards which recorded this record flood. The stage at the site was about 916 ft msl for an estimated flow of 51,000 ft^3/sec. Figure 2.4-3 shows the results of a flood frequency study. The 1000 year flood has an estimated stage of 921 ft msl. Since the plant grade is 930 ft msl, no flood problems are expected. 000-00

A study was made by the Harza Engineering Company to determine the predicted flood discharge flow and flood level at the site resulting from the maximum probable flood as defined by the U.S. Army Corps of Engineers (Policies and Procedures Pertaining to Determination of Spillway Capacities and Freeboard Allowances for Dams, Engineer Circular No. 1110-2-27, Enclosure 2, August 1, 1966 (Reference 33), Department of the Army, Office of the Chief of Engineers). Refer to Appendix G.

The probable maximum discharge was determined to be 364,900 ft^3/sec and to have a corresponding peak stage of elevation 939.2 ft msl. The flood would result from meteorological conditions which could occur in the spring and would reach maximum river level in about 12 days. It was estimated the flood stage would remain above elevation 930.0 ft msl. for approximately 11 days. Using this data, a study (See Section 12.2.1.7) was performed to identify flood protection requirements.

The normal river stage at the plant site is about 905 ft msl. At a distance 1-1/2 mile upstream, the normal river elevation is about 910 ft msl, and at an equal distance downstream, the river is at 900 ft msl. Thus, the hydraulic slope is about 3-1/3 ft/mile.

2.4.2 Public Water Supplies

2.4.2.1 Surface Water

The nearest domestic water supply reservoir with a free surface open to the air is the Minneapolis Water Works Reservoir. This reservoir is located north of Minneapolis, and is about 37 miles from the site. St. Paul uses a chain of lakes in its water supply system. These lakes, located north of St. Paul, are about 40 miles from the site.

The major supply of water for these reservoirs is the Mississippi River. The St. Paul intake is about 33 river miles from the site and the Minneapolis intake is about 37 miles from the site. Harza Engineering Company made a study of pollutant dispersion of a slug waste in the river (Reference 35) between the Monticello Plant site and the Minneapolis and St. Paul water intakes. The results of this study were given in Answer to Question 3.3 of Amendment 4 and all of Amendment 8 of the Monticello Facility Description and Safety Analysis Report.

In the event of a contaminated Mississippi River, the Minneapolis water supply would be more critical than the St. Paul water supply, because Minneapolis has about a 2 day water supply and St. Paul a 4+week supply. Under the emergency, withdrawal of river water for the Minneapolis system could be suspended for about 48 hours without curtailment of non-essential use. This period could be extended to about 100 hours if non-essential use is curtailed.

Between 1960 and 1980, recreational use of the reach of river near Monticello has increased significantly.

River water is used for irrigation in a limited way between the site and Minneapolis. Twenty-six water appropriation permits have been issued by the Minnesota Department of Natural Resources for this reach of the river.

At Elk River, the river water is used for cooling purposes for an electric generating plant. The next industrial water user is Xcel Energy in north Minneapolis. 01-010

2.4.2.2 Ground Water

The outwash drift on both sides of the Mississippi in general yields large quantities of water. The water table under normal circumstances is higher than the river, thus ground water as well as run-off from rainfall feeds the river. The drift water usually is quite hard containing calcium, magnesium, and bicarbonates, with small amounts of sodium, potassium, sulfates, and chlorides. Between the plant site and Minneapolis, the cities of Monticello, Elk River, Anoka, Coon Rapids, Champlin, Brooklyn Center, Brooklyn Park, and Fridley obtain groundwater from the bedrock formations for their domestic water supply as of 1981.

Numerous shallow wells supply water for residences and farms along the river terrace.

The closest public water supply well is the city of Monticello wells. These wells are 16 inches in diameter and 250 feet deep. The 1200 gpm capacity is limited by the installed pumps. The wells have been tested to 2000 gpm. They are located in the main part of the city of Monticello.

The wells which obtain their water from the drift are recharged by local precipitation, while the wells which withdraw water from the bedrock are recharged by precipitation where the bedrock is at or near the land surface. The largest increment of recharge occurs during the spring thaw.

At the plant site, the groundwater table was measured at about 922 ft msl. Since the normal river is at about 905 ft msl, groundwater flow is to the river. This usual case of groundwater flow to the river may not exist during floods.

2.4.3 Plant Design Bases Dependent on Hydrology

Water movements passing the site are subject to large variations in the course of a year. Plant design with respect to operation and liquid waste disposal takes into account large variations in water flow from less than 200 ft^3/sec to flood level up to plant grade (about 930ft msl) which is well above record historical floods. A study (see Section 12.2.1.7) was performed to identify flood protection requirements for floods up to the probable maximum flood (939.2ft msl).

2.4.4 Water Use Permits and Appropriations Relevant to Plant Operation

The ground and surface water appropriations are pursuant to permits issued by the Minnesota Department of Natural Resources. The requirements for groundwater include domestic use for over 25 persons, industrial use to seal pumps in the plant intake structure and plant make up water. River water is required for condenser cooling, service water cooling, and plant makeup.

2.4.5 Surface Water Quality

Water samples were taken upstream, downstream and at the plant discharge on February 28, 1972. The chemical analyses of the samples were as follows:

	Upstream Mississippi	Downstream Mississippi	Plant Discharge
P Alkalinity - ppm $CaCO_3$	0	0	0
M Alkalinity - ppm $CaCO_3$	170	169	165
Ammonia Nitrogen - ppm N	0.05	0.02	0.02
Organic Nitrogen - ppm N	0.933	0.61	0.65
Nitrate Nitrogen - ppm N	0.28	0.37	0.37
Nitrite Nitrogen - ppm N	0.001	0.003	0.002
Chloride - ppm	1.4	0.9	1.0
Sulfate - ppm SO_4	7.8	6.6	7.3
Color - Units	35	35	35
Turbidity - JTU	3.9	2.0	2.5
Total Hardness - ppm $CaCO_3$	177	178	178
Calcium Hardness - ppm $CaCO_3$	122	114	122
pH	7.5	7.9	7.8
Total Solids - ppm	288	272	247
Non-Filterable Solids - ppm	12	3	5
Dissolved Solids - ppm	276	269	242

	Upstream Mississippi	Downstream Mississippi	Plant Discharge
Fixed Non-Filterable Solids - ppm	8	2	3
Volatile Solids - ppm	4	1	2
Total Soluble Phosphorus - ppm P	0.035	0.026	0.024
Total Chlorophyll - mg/m^3	5.7	1.5	1.6
Conductivity - mmhos (25°C)	364	357	364
Temp. °C	0.2	8.3	15.5
D.O. mg/l	8.4	8.6	8.2
BOD mg/l	0.9	1.0	0.9

Cooling towers not operating

Both paper pulp (Sartell and Little Falls) and sewage treatment facilities (St. Cloud and others) are located upstream of the plant.

Table 2.4-1 Mississippi River Flows at Elk River and St. Cloud, Minnesota

	Location	
	Elk River[1]	St. Cloud[2]
Number of Records, years	38	40
Average Annual Flow, ft^3/sec	5,260	4,360
Minimum Recorded Flow, ft^3/sec	278	220
Maximum Recorded Flow, ft^3/sec	49,200	46,780
	(4-12-52)	(4-15-65)

1. Data from Hydrologic Atlas of Minnesota, Bulletin #10, Minnesota Department of Conservation, April 1959, at U.S. Geological Survey, Recorder 2755. Station discontinued October 31, 1957 (Reference 36).

2. Data from Northern States Power Company records from July 1, 1925, to December 31, 1965, at Whitney Steam Plant, St. Cloud, Minnesota (Reference 37).

2.5 Geology and Soil Investigation

2.5.1 General

Dames and Moore, consultants in applied earth sciences, analyzed the geology and foundation conditions of the plant site.

2.5.2 Regional Geology

Rocks dating as early as Precambrian time underlie the region of Minnesota which includes the plant site. Pleistocene glaciation, probably less than 1,000,000 years in age, as well as recent alluvial deposition have mantled the older rocks with a variety of unconsolidated materials in the form of glacial moraines, glacial outwash plains, glacial till, and river bed sediments. This cover of young soil rests upon a surface of glacially-carved bedrock consisting of sandstone and shale strata underlain by deeply weathered granite rocks. Volcanics also form portions of the bedrock sequence in certain areas. The bedrock surface is irregular and slopes generally to the east or southeast.

The geologic column showing the age relationships of the various bedrock units and surficial deposits of the region is presented in Table 2.5-1. Figure 2.5-1a and 2.5-1b show the regional extent of the consolidated formations.

The principal structural feature in this part of Minnesota is a deep trough formed during Precambrian time in the granite and associated crystalline rocks. This basin extended from Lake Superior into Iowa, and provided a site for the deposition of thick sequences of Precambrian and later Paleozoic sediments and volcanics. Strata of Paleozoic age are now exposed along the southern half of the structural trough. In the Minneapolis-St. Paul area, they form a circular basin containing artesian groundwater.

The ice fronts or glacial lobes advanced across this region during the last stage of glaciation, named the Wisconsin Stage. One lobe came from the general area of Lake Superior and deposited terminal moraines immediately south of the present course of the Mississippi River. A later ice front advanced across the area from the southwest, overriding the earlier moraines. Erosion of these glacial sediments by the Mississippi River has been active since the final retreat of the ice.

The present course of the Mississippi has no relation to the streams that flowed through the area prior to glaciation. There are therefore, old river channels which cross the region and which may be substantially deeper than the present river channel.

A major fault system of Precambrian age has been inferred from regional geophysical surveys. This fault system is associated with the Precambrian structural trough. The major movements along this fault system, which amount to thousands of feet, appear to have been restricted to Precambrian time. Minor fault displacements occurred during the Paleozoic era, but faulting within the last few million years is not in evidence.

2.5.3 Site Geology

The site occupies a bluff which forms the southwest bank of the Mississippi River. Several flat alluvial terraces comprise the main topographical features on the property. These terraces lie at average elevations of 930 and 918 ft msl and in general, slope very slightly away from the river.

The present surface drainage of the immediate plant site area is mainly to the southwest, away from the river. Surface run-off will tend to collect in the depression at the south end of the terrace where it is bounded by higher ground, then flow easterly to the river.

At the time of start of construction, most of the site was under cultivation, which has since been discontinued, with the remainder of the site area covered by scattered low brush and small trees.

The pattern of the present meander system suggests that the channel to the south of the islands in the river is now the main channel. It is possible that the channel to the north of the islands may eventually be abandoned. If this occurs during the lifetime of the plant it probably will result in increased erosion along the bluff at the plant site; however, this erosion is not a matter of concern because the actual amount would be small and not interfere with any structures.

The site is located on the extreme western edge of the Precambrian structural trough previously discussed under Regional Geology. A well in the town of Monticello about 2-3/4 miles east of the site which was drilled to a depth of 500 ft did not encounter granite. Other well information generally indicates that 150 to 200 ft of unconsolidated alluvium and drift overlies sandstone and red shale of unknown thickness at Monticello. All the rock and soil units present at the site therefore slope eastward and thicken toward the sedimentary basin and its artesian groundwater aquifers.

Decomposed granite and basic rocks of Precambrian age comprise the oldest formation at the site, within the depth investigated. This material lies below the ground surface at a depth of about 75 to 122 ft. (See Figures 2.5-1a through 2.5-5) Resting directly upon the weathered Precambrian crystalline rocks is approximately 10 to 15 ft of medium-grained quartz sandstone which, in general, is moderately well cemented. The upper surface of underlying rock can support unit foundation loads up to 15,000 pounds per square foot.

Above the sandstone is a series of alluvial strata about 50 ft thick which consists predominately of clean sands with gravel, as well as a few layers of clay and glacial till. This alluvial sequence represents successive depositions of glacial outwash, moraine, and more recently, sediments laid down by the Mississippi River. During its history this river has meandered as much as 1-1/2 miles south of its present channel.

The distribution of the unconsolidated materials in the locality of the site is shown on Figure 2.5-1b.

The nearest known or inferred fault is the Douglas fault, located approximately 23 miles southeast of the site as shown on Figure 2.5-1a. It is probable that the site has not experienced any activity within recent geologic times.

2.5.4 Groundwater

Large supplies of groundwater are available from the Mississippi River sediments, the glacial deposits, and the underlying sandstones in the area. Most of the private wells in the area are shallow, and penetrate either the river alluvium or the glacial deposits. The town of Monticello derives its water supply from a well approximately 237 ft deep which is believed to penetrate sandstone aquifers. The communities of Big Lake, Albertville, and Elk River also recover water from this formation.

The general path of deep groundwater flow is to the southeast across the region surrounding the site for the plant. The regional gradient, therefore, broadly parallels the trend of the topography and the principal surface drainage. Groundwater at shallower depths moves toward the Mississippi River or its tributaries at variable gradients depending on local conditions.

The water table beneath the low terraces which border the Mississippi River usually lies at about river elevation and slopes very slightly toward the river during periods of normal stream flow. Such is the case at the site.

Movement of groundwater takes place within the three principal rock and soil materials at the site. In the decomposed, clayey granitic rocks, which are very low in permeability relative to the overlying materials, the rate of ground water movement is extremely slow.

2.5.5 Foundation Investigation

The location of the principal structures including the turbine and reactor buildings, intake structure, stack and diesel building and soil borings are shown in Figures 2.5-1a through 2.5-5.

Dynamic soil tests were not considered because the probability of liquefaction is very low under the cyclic loadings produced by the 1952 Taft earthquake (refer to Section 2.6.3), considering the density of the sand and overburden pressure.

Sands which are typically vulnerable to liquefaction are saturated, under low confining pressures, and have standard penetration test values of about N=5. Laboratory studies by Seed and Lee (Liquefaction of Saturated Sands during Cyclic Loading, Journal Soil Mechanics and Foundation Division, ASCE, November 1966, Volume 92, No. SM6) (Reference 38) demonstrate that sands denser than the critical void ratio can be made to liquefy under cyclic loading. Consequently liquefaction has an extremely low statistical possibility in a cemented sand with standard penetration test values of N=80 or more, and could only occur under a very large number (e.g., 10,000) of very high stress cycles. The number of stress cycles that could be expected due to the Taft earthquake is estimated to be less than 1000 cycles.

2.5.6 Conclusions

No unusual features of the site geology are evident. Underlying formations are adequate for foundation for the plant structures.

The geology and soil conditions have been investigated and found stable. Consequently, no special plant design features pertaining to the site geology were necessary.

Table 2.5-1 Geologic Formations in the General Area of the Site

Geologic Age: ERA	Geologic Age: Period	Geologic Name	Description	Remarks
Cenozoic	Quaternary	Recent Deposits	Unconsolidated clay, silt, sand, and gravel	Largely Mississippi River deposits
		Pleistocene	Unconsolidated clay, silt, sand, gravel, and boulders deposited as till, outwash, lake deposits, & loess	Largely from Superior and Grantsburg lobes of Wisconsin glaciation
Paleozoic	Cambrian	Franconia Formation (St. Croix Series)	Sandstone and shale, some aquifer zones	May not be present in immediate area of site
		Dresbach Formation (St. Croix Series)	Sandstone, siltstone and shale, aquifer zone	May not be present in immediate area of site
Precambrian	Keweenawan	Hinckley Formation	Sandstone	Thin in the immediate area of the site. An important aquifer where sufficiently thick
		Red Clastic Series	Sandstone and red shale	Probably not present in immediate area of site
		Volcanics	Mafic lava flows with thin layers of tuff and breccia	Probably not present in immediate area of site
		Granite and Associated Intrusives		Present at site

2.6 Seismology

2.6.1 General

John A. Blume, Associates, analyzed the seismology of the plant site. A copy of the Blume report is included in Appendix A.

2.6.2 Seismic History

In Table 2.6-1 are listed numerically the earthquakes in the general region in and around Minnesota. Those more applicable to the site are plotted on Figure 2.6-1. The earliest earthquake on record occurred in 1860 in central Minnesota; thus over 100 years of records exist. During that period, earthquakes have had little effect at the site. Since compilation of Table 2.6-1, there has been no observed evidence of seismic activity in the plant area.

2.6.3 Faulting in Area

The nearest known or inferred fault - the Douglas Fault - is 23 miles southeast of the site (Figure 2.5-1a). According to referenced geological information, there is no indication that faulting has affected the area of the site in the last few million years. The major fault system of Precambrian age, which is associated with the Precambrian structural trough, is seen on Figure 2.6-2. Major movements of thousands of feet along this system appear to have been restricted to Precambrian time, with minor displacements having occurred during the Paleozoic era. Faulting within recent geologic time is not in evidence.

Richter's Seismic Regionalization Map (Figure 2.6-3) shows the area of the site in a probable maximum intensity of VIII, Modified Mercalli.

This intensity has been based on the area's relationship to the Canadian shield. Stable shields in other continents are usually fringed by belts of moderate seismicity, with occasionally large earthquakes. Historically, this area is too young to prove or disprove such seismic activity. The Modified Mercalli scale is explained in Table 2.6-2.

The Coast and Geodetic Survey's Seismic Probability Map of the United States (Figure 2.6-4) assigns the area to Zone 0 - no damage.

It is considered that neither the regionalization nor the probability map is satisfactory in determining a proper seismic factor if considered alone. Each, however, is based on judgment and fact which, when weighed with other data, become more meaningful. In the case at hand, the assignment of an VIII as the largest probable intensity for the general area must be tempered by the fact that the intensity at or near the underlying sandstone will be much less than that experienced in areas of less competent material, where invariably the maximum damage is sustained.

Earthquakes can and do occur in this region away from faults, and probably result from residual stresses due to recent glaciers. A quake similar to No. 12 and 24 in Table 2.6-1 was postulated near the site and using the dynamic response data obtained insitu, the Taft earthquake of July 21, 1952, North 69 West component with an applied factor of 0.33 was selected as best representative for the design earthquake. Figure 2.6-5 shows single-mass spectra when averaged.

2.6.4 Design Criteria

Design criteria which utilize this earthquake record are discussed in Section 12. Section 12 also gives specific design information related to the seismic analysis of the building and equipment.

2.6.5 Seismic Monitoring System

The Seismic Monitoring System annunciates the occurrence and records the severity of significant seismic events.

The system is composed of three subsystems: the relatively simple annunciators and peak-recording accelerometers, and the more sophisticated acceleration sensors located in the drywell, on the refueling floor and in the seismic shed (located to the north of the warehouse).

Each of the peak-recording accelerometers is a self-contained unit. The sensing mechanism is a permanent magnet stylus attached to the end of a torsional accelerometer. Low frequency accelerations cause the magnet to erase pre-recorded lines on a small (approximately 1/4 inch square) piece of magnetic tape. Because an erasure is permanent, only the peak acceleration that the tape has been subjected to can be deduced when the tape is developed. Each peak recording accelerometer unit contains three torsional accelerometers and magnetic tapes - one each for longitudinal, transverse, and vertical accelerations.

The magnetic tapes can be removed from the accelerometers, developed, and evaluated by plant personnel for a rapid determination of the severity of a seismic disturbance.

The accelerograph recording system gives a more detailed record of a disturbance than the peak recording accelerometers - it records accelerations in three directions (longitudinal, transverse, and vertical, as above) at each of the three sensor locations on magnetic tape cartridges. This system has five major components: trigger, three sensors, and the recording and control equipment. When the trigger (located in the No. 12 125 Vdc battery room) senses the beginning of a seismic disturbance, (an acceleration $\geq$.01 g), it initiates the system power-on sequence and causes the EARTHQUAKE alarm to annunciate in the control room. The recorder then converts the nine analog acceleration signals (three sensors with three directions/sensor) into frequency modulated tones and records them on the magnetic tapes (one for each triaxial sensor). The recorder will run for 10 seconds after each trigger signal, up to a maximum of 30 minutes. The resulting tape gives a detailed record of the disturbance, but must be sent off-site to be fully processed.

The control room EARTHQUAKE annunciator is also initiated by any seismic switch of the Seismic Annunciator System. In addition to this, there are two more alarms initiated by the Seismic Annunciator System. The first of these is the Operational Basis Earthquake (OBE) alarm which annunciates when its seismic switch senses an acceleration $\geq$.03g. The second is the Design Basis Earthquake (DBE) alarm, which annunciates when its switch senses an acceleration $\geq$.06g. These two switches do not activate the accelerograph recording system.

Table 2.6-1 Seismic History of the Region

(Page 1 of 2)

No.	Date	Location: Place	Location: N Lat.	Location: W Long.	Intensity (M.M.)	Remarks
* 1	1860	Central Minn.	-	-	Unknown	Felt over 3,000 square miles
2	10/9/1872	Sioux City, Iowa	42.7	97.0	V	Felt over 140,000 square miles
3	11/15/1877	East Nebraska	41.0	97.0	VII	Felt over 140,000 square miles.
4	7/28/1902	East Nebraska	42.5	97.5	V	Felt over 35,000 square miles.
5	7/26/1905	Calumet, Mich.	47.3	88.4	VII	Felt over 16,000 square miles.
6	5/9/1906	Washabaugh County, S. D.	43.0	101.0	VI	Felt over 8,000 square miles.
7	5/26/1906	Keweenaw Peninsula, Michigan	47.3	88.4	VIII	Felt over 1,000 square miles.
8	5/15/1909	Canada, felt to South	50.0	105.00	VIII	Felt over 500,000 square miles.
9	5/26/1909	Dixon, Illinois	42.5	89.0	VII	Felt over 40,000 square miles.
10	10/22/1909	Sterling, Illinois	41.6	89.8	IV-V	
11	6/2/1911	South Dakota	44.2	98.2	V	Felt over 40,000 square miles.
12	9/3/1917	Minnesota	46.3	94.5	VI	Felt over 10,000 square miles.
*13	2/28/1925	Canada	48.2	70.8	VIII	Felt over 2,000,000 square miles.
14	10/6/1929	Yankton, S. Dakota	42.8	97.4	V (est.)	
15	1/17/1931	White Lake, S. Dakota	43.8	98.7	V (est.)	
*16	11/12/1934	Rock Island & Moline, Illinois Davenport, Iowa	41.4	90.5	V	
17	3/1/1935	Eastern Nebraska	40.3	96.2	VI	Felt over 50,000 square miles.
*18	11/1/1935	Canada	46.8	79.1	IX and over	Felt over 1,000,000 square miles, felt in Minnesota.
19	11/1/1935	Egan, S. Dakota	44.0	96.6	V (est.)	
20	10/1/1938	Sioux Falls, S. Dakota	43.5	96.6	V	Felt over 3,000 square miles.

* Indicates epicenter not plotted on map.

Table 2.6-1 Seismic History of the Region

(Page 2 of 2)

No.	Date	Location: Place	Location: N Lat.	Location: W Long.	Intensity (M.M.)	Remarks
21	1/28/1939	Detroit Lake, Minn.	46.9	95.5	V (est.)	
22	6/10/1939	Fairfax, S. Dakota	43.1	98.8	VI (est.)	
23	7/23/1946	Wessington, S. Dakota	44.5	98.7	VI (est.)	
24	5/6/1947	Milwaukee Area	42.9	87.9	VII	Felt Sheboygan to Kenosha, Wis.
25	2/15/1950	Alexandria, Minn.	45.7	94.8	V-VI (est.)	
26	1/6/1955	Hancock, Michigan	47.3	88.4	V	
27	12/3/1957	Mitchell, S. Dakota	43.8	98.0	V	
28	1/12/1959	Doland, S. Dakota	44.9	98.0	V	
29	12/31/1961	W. Pierre, S. Dakota	44.4	100.5	VI	

Table 2.6-2 Modified Mercalli Intensity Scale of 1931 (Abridged)

I. Not felt except by a very few under especially favorable circumstances

II. Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing.

III. Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing of truck. Duration estimated.

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed, walls make creaking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably.

V. Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbance of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop.

VI. Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight.

VII. Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motor cars.

VIII. Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Disturbs persons driving motor cars.

IX. Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken.

X. Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks.

XI. Few, if any (masonry), structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipe lines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.

XII. Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into the air.

2.7 Radiation Environmental Monitoring Program (REMP)

2.7.1 Program Design and Data Interpretation

The purpose of the Radiation Environmental Monitoring Program (REMP) at the Monticello Nuclear Generating Plant is to assess the impact of the plant on its environment (References 7 and 42). For this purpose, samples are collected from the air, terrestrial, and aquatic environments and analyzed for radioactive content. In addition, ambient gamma radiation levels are monitored by thermoluminescent dosimeters (TLDs).

Sources of environmental radiation include the following:

a. natural background radiation arising from cosmic rays and primordial radionuclides;

b. fallout from atmospheric nuclear detonations;

c. releases from nuclear power plants.

In interpreting the data, effects due to the Plant must be distinguished from those due to other sources. To accomplish this, the program uses the control-indicator concept suggested by NRC Guidelines.

2.7.2 Program Description

The sample types and locations included in the current Radiation Environmental Monitoring Program (REMP) at the Monticello Nuclear Generating Plant are listed in the Offsite Dose Calculation Manual (ODCM, Reference 8).

Sample locations are chosen to provide measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures off site. The technique for establishing sample locations conforms to guidance provided by the NRC.

The air environment is monitored by continuous air samplers which filter out airborne radioactive particulates and adsorb airborne radioiodine.

Ambient gamma radiation is monitored at thermoluminescent dosimeter (TLD) stations located in a circular array around the plant.

The terrestrial environment is monitored through samples of groundwater and locally produced food products.

The aquatic environment is monitored through sampling sediment and water from the Mississippi River at locations upstream and downstream of the plant. Drinking water from the city of Minneapolis, which is drawn from the river, is also sampled.

2.7.3 Interlaboratory Comparison Program

Monticello participates in an Interlaboratory Comparison Program to ensure the precision and accuracy of radioactivity measurements of environmental samples. This program is described in the ODCM.

2.8 Ecological and Biological Studies

On August 26, 1977 the Minnesota Pollution Control Agency, the permitting agency under the U. S. Environmental Protection Agency, issued the National Pollution Discharge Elimination System (NPDES) Permit No. MN0000868 covering the Monticello Nuclear Generating Plant. This permit is reissued with any modifications required every 5 years. The NPDES effluent limitations and monitoring requirements, thermal studies and ecological monitoring requirements provide appropriate protection for the environment. There are no ecological or biological monitoring requirements under NRC jurisdiction. Pre-operational and early operational ecological and biological studies are described in the FSAR.

FOR ADMINISTRATIVE USE ONLY					
ɘsp Supv: TOPE		Assoc Ref: USAR-MAN		SR: N	Freq: 2 yrs
ARMS: USAR-02.08	Doc Type: 9703		Admin Initials:		Date: 01/31/03

l/djm

2.9 Consequences of Hypothetical Local Catastrophes

2.9.1 Toxic Chemical Spills

Due to the toxicity of commonly used chemicals, which may be transported near the Monticello Nuclear Generating Plant by railroad or highway, a survey was performed to predict which chemicals may become hazardous in the event of a spill. The analysis was performed in conformance with the guidance set forth by Regulatory Guide 1.78 (Reference 40) and NUREG 0570 (Reference 41). The analysis results were submitted to the NRC for review as required by NUREG 0737, Item III D.3.4 (References 10, 11, 12, 13).

A new toxic chemical survey (Reference 16) was performed in 1993 which identified toxic chemicals in sufficient quantities stored on-site, stored in the vicinity of the site, or shipped near the plant at sufficient frequency to warrant further evaluation. For chemicals meeting these criteria, evaluation indicated that Control Room personnel would have at least two minutes to don breathing apparatus before incapacitation limits were exceeded. The results of the 1993 survey and evaluation were submitted (References 17 and 43) and approved by the NRC (Reference 44).

In 1998, the list of postulated spills was reviewed. The 1993 methodology was used, with updated Control Room air intake rates and volume, to determine event duration. These event durations were then used to size the Control Room Breathing Air System (see Section 10.3.11).

2.10 References

1. Deleted.
2. "Tornado Probabilities", H C S Thom, Office of Climatology, U.S. Weather Bureau, published in Monthly Weather Review, October-November 1963, 91, Nos. 10-12, Pages 730-736.
3. General Electric Report, APED-5696, "Tornado Protection for Spent Fuel Storage Pool", D R Miller and W A Williams, November 1968.
4. "Air Pathway Exposure Model Validation Study at the Monticello Nuclear Generating Plant", J E Partridge, et al., Eastern Environmental Radiation Facility, Montgomery, Alabama and C B Nelson, Environmental Analysis Division, U.S. Environmental Protection Agency.
5. General Electric Report, NEDO-21159-1, "Airborne Releases from BWR's for Environmental Impact Evaluations, Amendment 1", T R Marrero, September 1976.
6. NUS Report, "Monticello Short Term (Accident) Diffusion Factors", D A Sullivan, Environmental Safeguards Division, Rockville, Maryland, August 1976.
7. NSP (L O Mayer) letter to the NRC, License Amendment Request - "Radioactive Effluent Technical Specification", dated May 1, 1979.
8. Monticello Nuclear Generating Plant, Offsite Dose Calculation Manual (ODCM).
9. NSP (L O Mayer) letter to the NRC (D L Zieman), "MNGP Appendix I Filing", dated June 4, 1976.
10. Bechtel Report, "Monticello Nuclear Generating Plant Main Control Room, Toxic Chemical Study", January 1981.
11. NSP (D M Musolf) letter to the NRC, "Revision 1 to License Amendment Request dated April 3, 1984", dated August 17, 1984.
12. NRC (J A Zwolinski) letter to NSP (D M Musolf), "Request for Additional Information - Control Room Ventilation System (TAC 56977)", dated November 5, 1986.
13. NSP (D M Musolf) letter to the NRC, "Revision 3 to License Amendment Request dated April 3, 1984", dated February 19, 1987.
14. NUREG/CR-2919, "XOQDOQ, Computer Program for the Meteorological Evaluation of Routine Effluent Release at Nuclear Power Stations", September 1982.
15. Earth Tech document, "Evacuation Time Estimates for the Monticello Power Plant Plume Exposure Pathway Emergency Planning Zone", December 1997.

16. Tenera Report, "Monticello Nuclear Generating Plant Control Room Habitability - Toxic Chemical Study", Final Report", June 11, 1993.

17. NSP (R O Anderson) letter to the NRC, "License Amendment Request, Removal of Chlorine Detection Requirements and Changes to Control Room Ventilation System Requirements", dated November 30, 1993.

18. Housing and Home Finance Agency (HHFA) study conducted in 1952.

19. American Telephone and Telegraph Company study, conducted 1917-18 to 1924-25.

20. Edison Electric Institute study, conducted 1926-27 to 1937-38.

21. Association of American Railroads study, conducted 1928-29 to 1936-37.

22. Quartermaster Research and Engineering Command Study, U.S. Army, 1959.

23. "Rainfall Intensity - Duration - Frequency Curves", U.S. Weather Bureau Technical Paper No. 25, (1955).

24. "Climatological Data with Comparative Data, Minneapolis - St. Paul, Minnesota, 1953-1956", U.S. Weather Bureau (2 publications).

25. "Climatological Data with Comparative Data, St. Cloud, Minnesota 1953 - 1965", U.S. Weather Bureau (2 publications).

26. "Climatography of the United States", No. 86-17, Minnesota, U.S. Weather Bureau.

27. "Local Climatological Data with Comparative Data, 1965", U.S. Weather Bureau.

28. "Snow Load Studies", Housing Research Paper 19, Housing and Home Finance Agency, 1952.

29. "Glaze, Its Meteorology and Climatology, Geographical Distribution and Economic Effects", Quartermaster Research and Engineering Center, 1959.

30. "Climatography of the United States", No. 60-21, Minnesota, U.S. Weather Bureau.

31. Mean Annual Tornado Frequency figures published by the U.S. Department of Commerce, Weather Bureau.

32. Deleted.

33. Department of the Army, Office of the Chief of Engineers, Engineer Circular No. 1110-2-27, "Policies and Procedures Pertaining to Spillway Capacities and Freeboard Allowances for Dams", August 1, 1966 .

34. Deleted.

35. Harza Engineering Company study of pollutant dispersion of a slug waste in the Mississippi River.

36. "Hydrologic Atlas of Minnesota", Bulletin #10, Minnesota Department of Conservation, April 1959, at U.S. Geological Survey, Recorder 2755.

37. Northern States Power Company records from July 1, 1925, to December 31, 1965, at Whitney Steam Plant, St. Cloud, Minnesota.

38. "Liquefaction of Saturated Sands during Cyclic Loading", Laboratory studies by Seed and Lee, ASCE, Journal, Soil Mechanics and Foundation Division, Volume 92, No. SM6, November 1966.

39. Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants", Proposed Revision 1, September 1980.

40. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During A Postulated Hazardous Chemical Release", June 1, 1974.

41. NUREG 0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release", June 1, 1979.

42. NRC (D B Vassallo) letter to NSP (D M Musolf), Amendment 15 to Facility Operating License DPR-22 dated December 17, 1982.

43. NSP (R O Anderson) letter to the NRC, "Additional Information to Support License Amendment Request dated November 30, 1993", dated June 30, 1994.

44. NRC (B A Wetzel) letter to NSP (R O Anderson), "Issuance of Amendment Re: Removal of Chlorine Detector Requirements from Technical Specifications (TAC No. M88602)", dated August 25, 1994.

FIGURES

FOR ADMINISTRATIVE USE ONLY				
Resp Supv: TOPE		Assoc Ref: USAR-MAN	SR: N	Freq: 2 yrs
ARMS: USAR-2.FIGURES	Doc Type: 9703	Admin Initials:	JH	Date: 10/13/08

l/jmr

Figure 2.3-1 Return Period of Extreme Short-Interval Rainfall, Minneapolis, MN

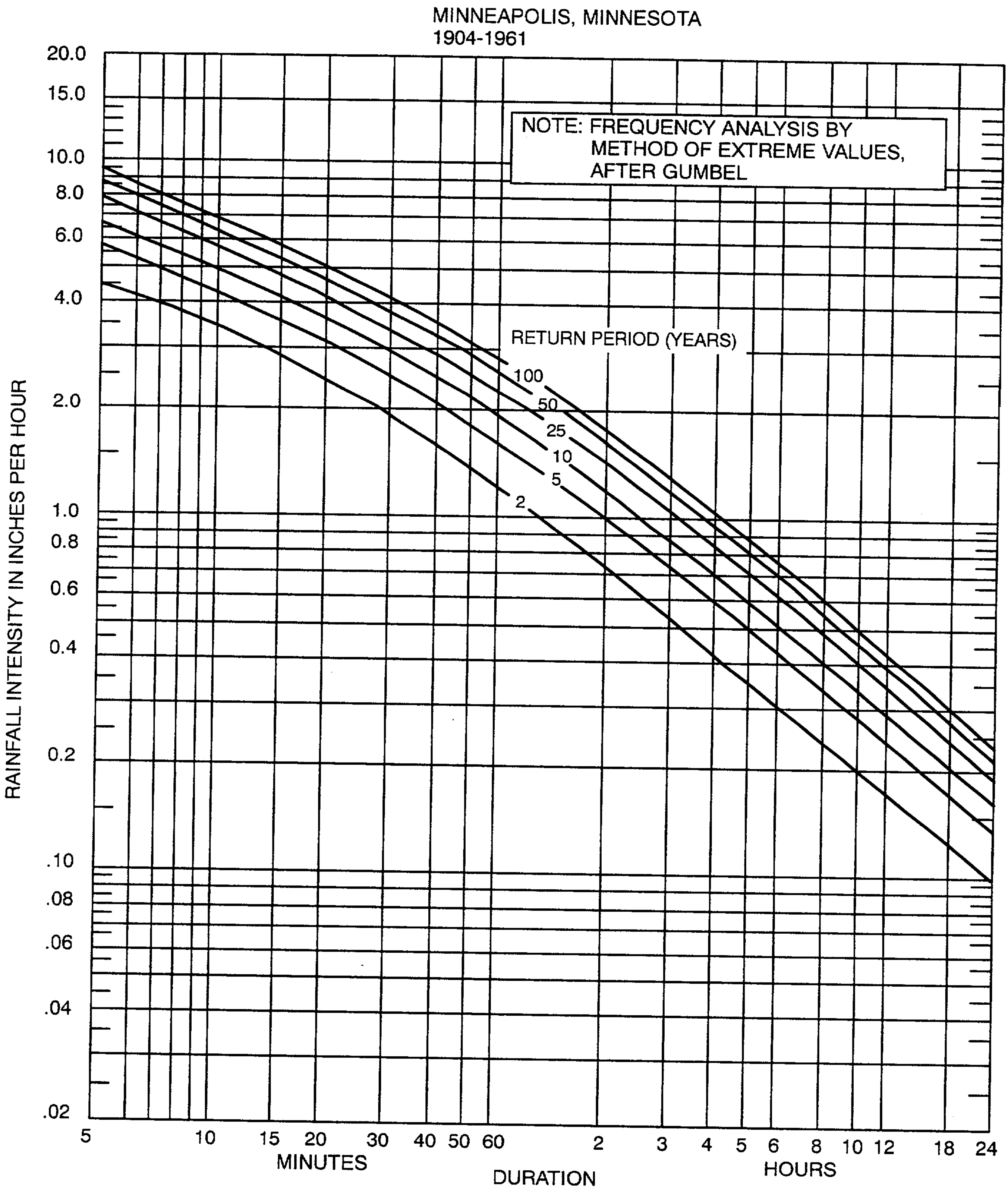


Figure 2.4-1 Flow Duration Curve, Mississippi River at St. Cloud, MN

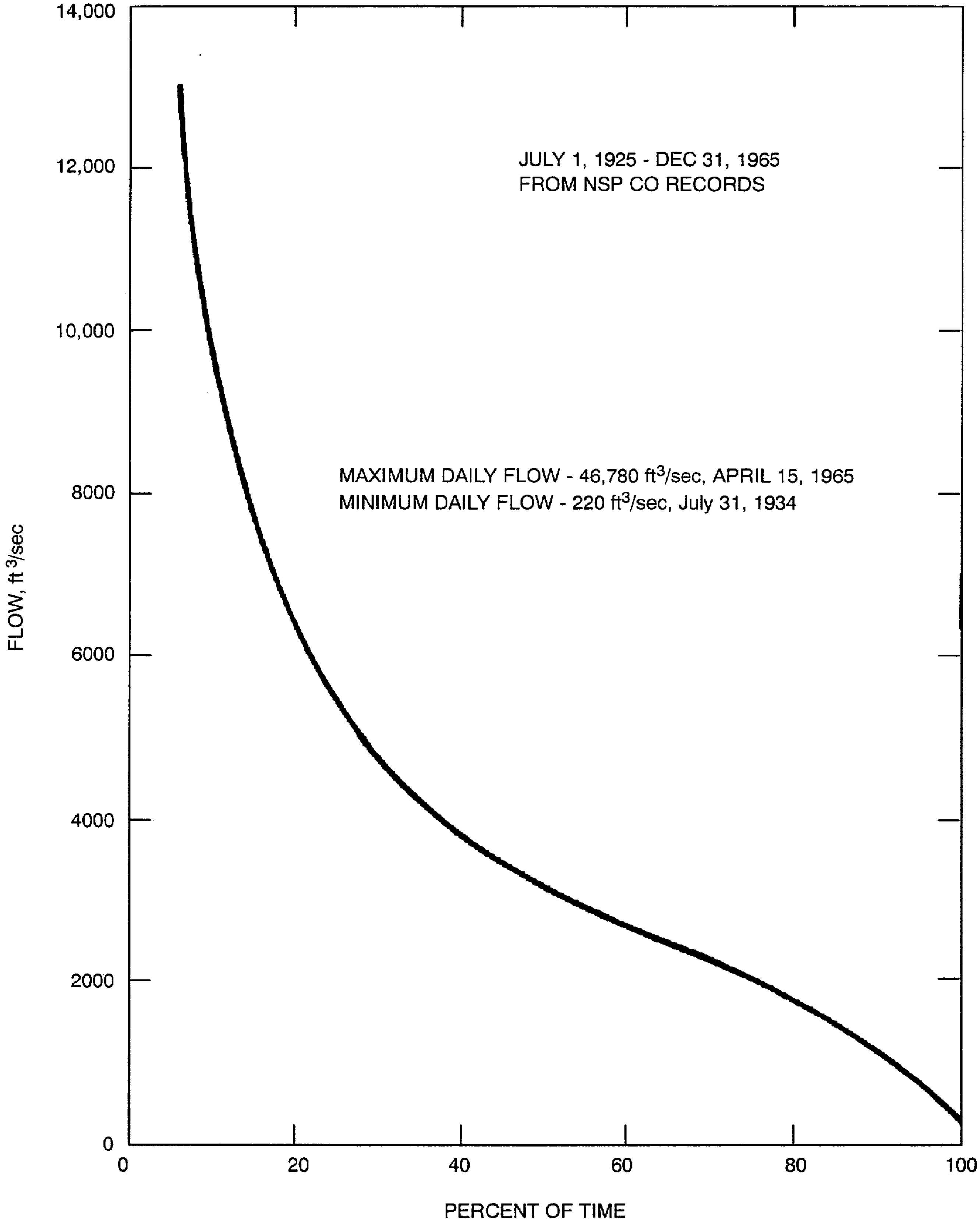


Figure 2.4-2 1965 Spring Flood at Monticello Site

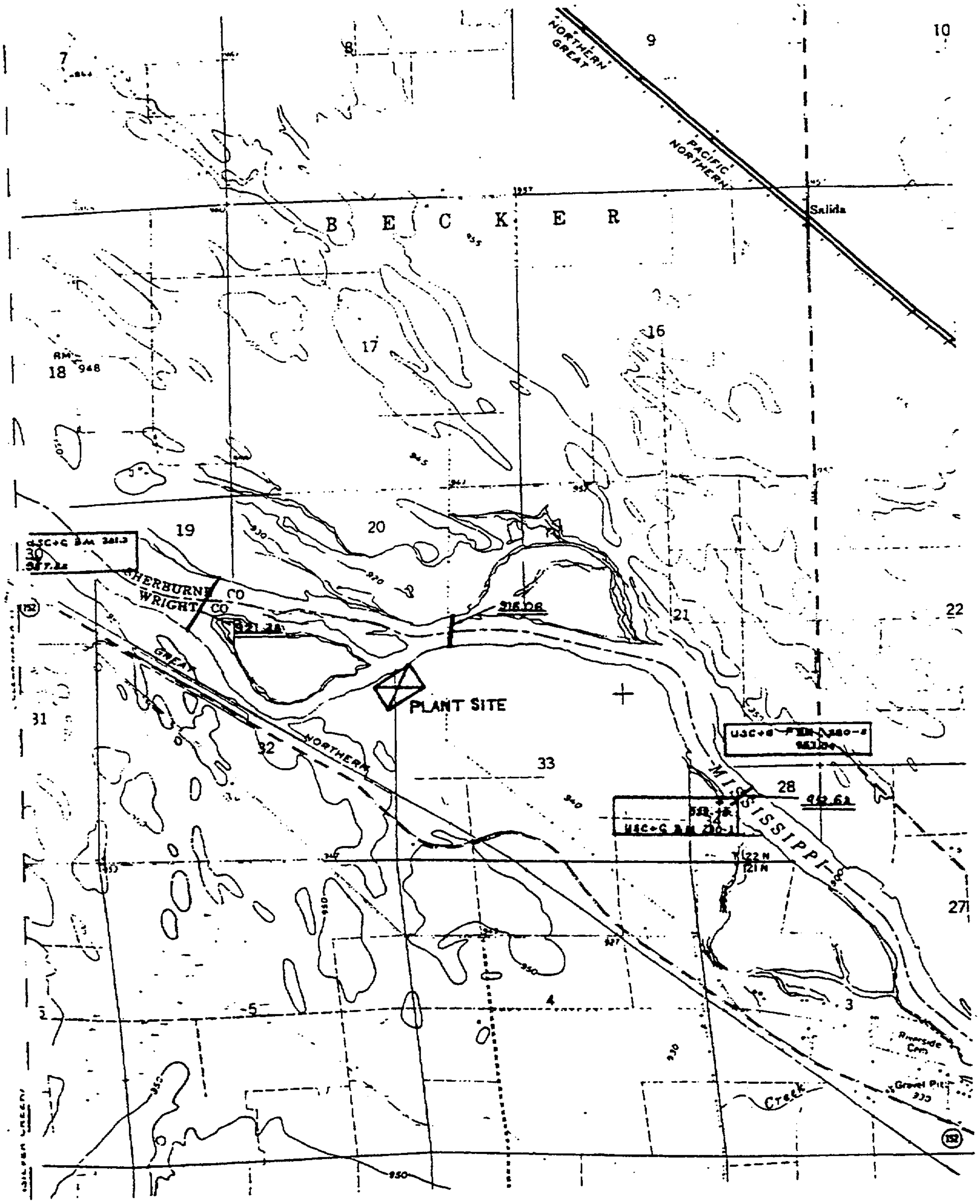


Figure 2.4-3 Flood Frequency Study - Mississippi River at Monticello Site

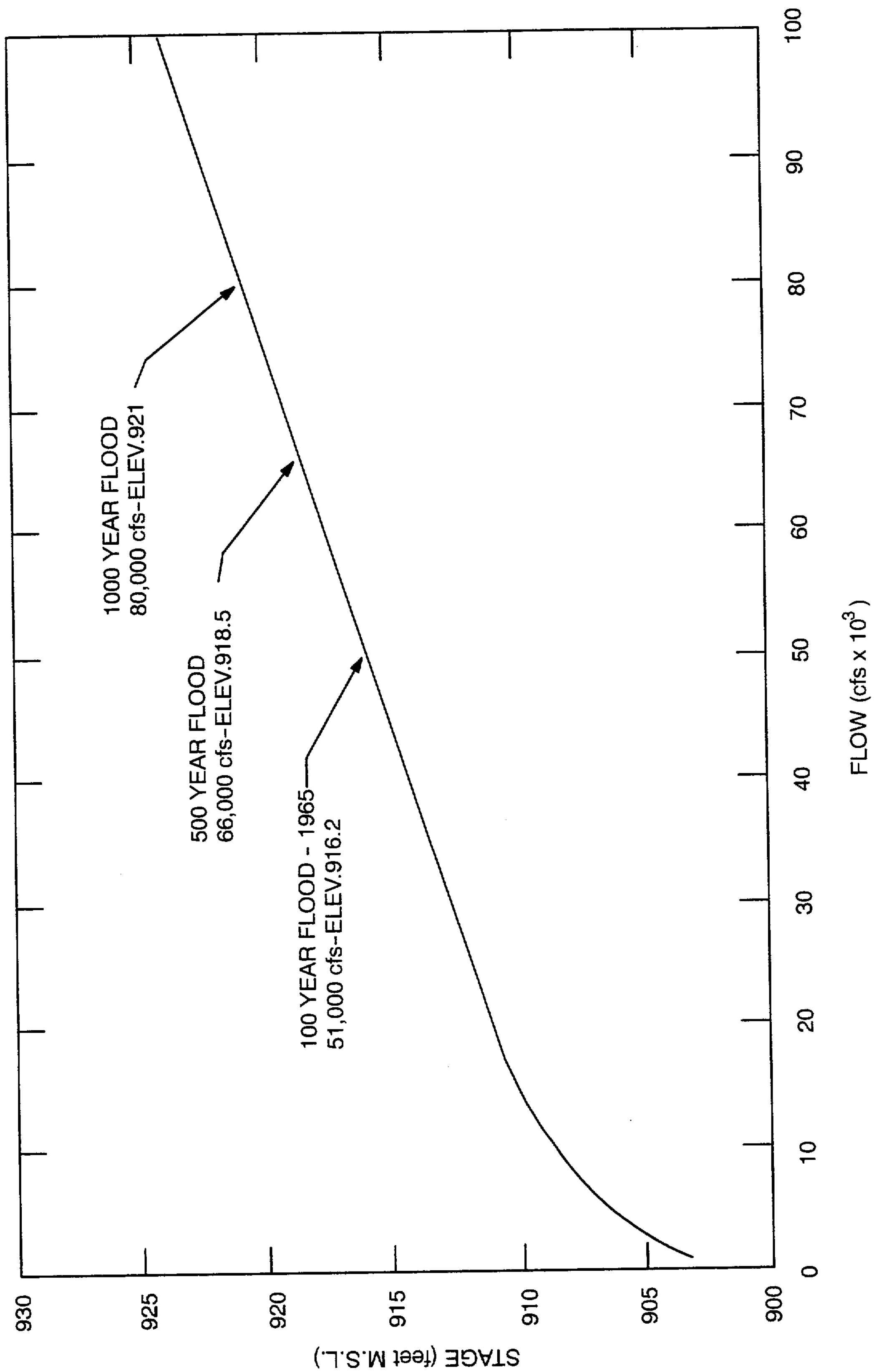


Figure 2.5-1a Overlay Regional Tectonic Map

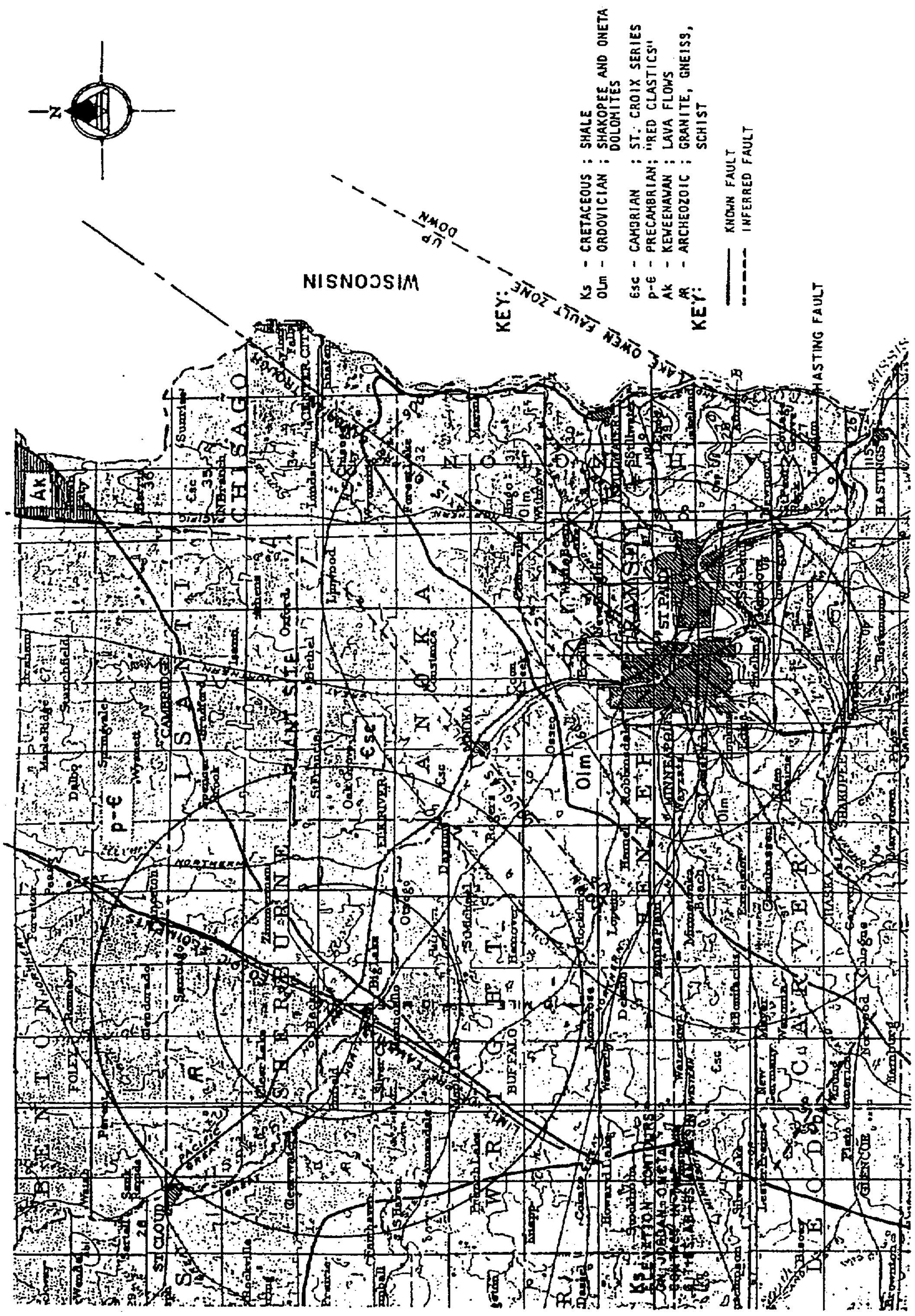


Figure 2.5-1b Regional Geology Map

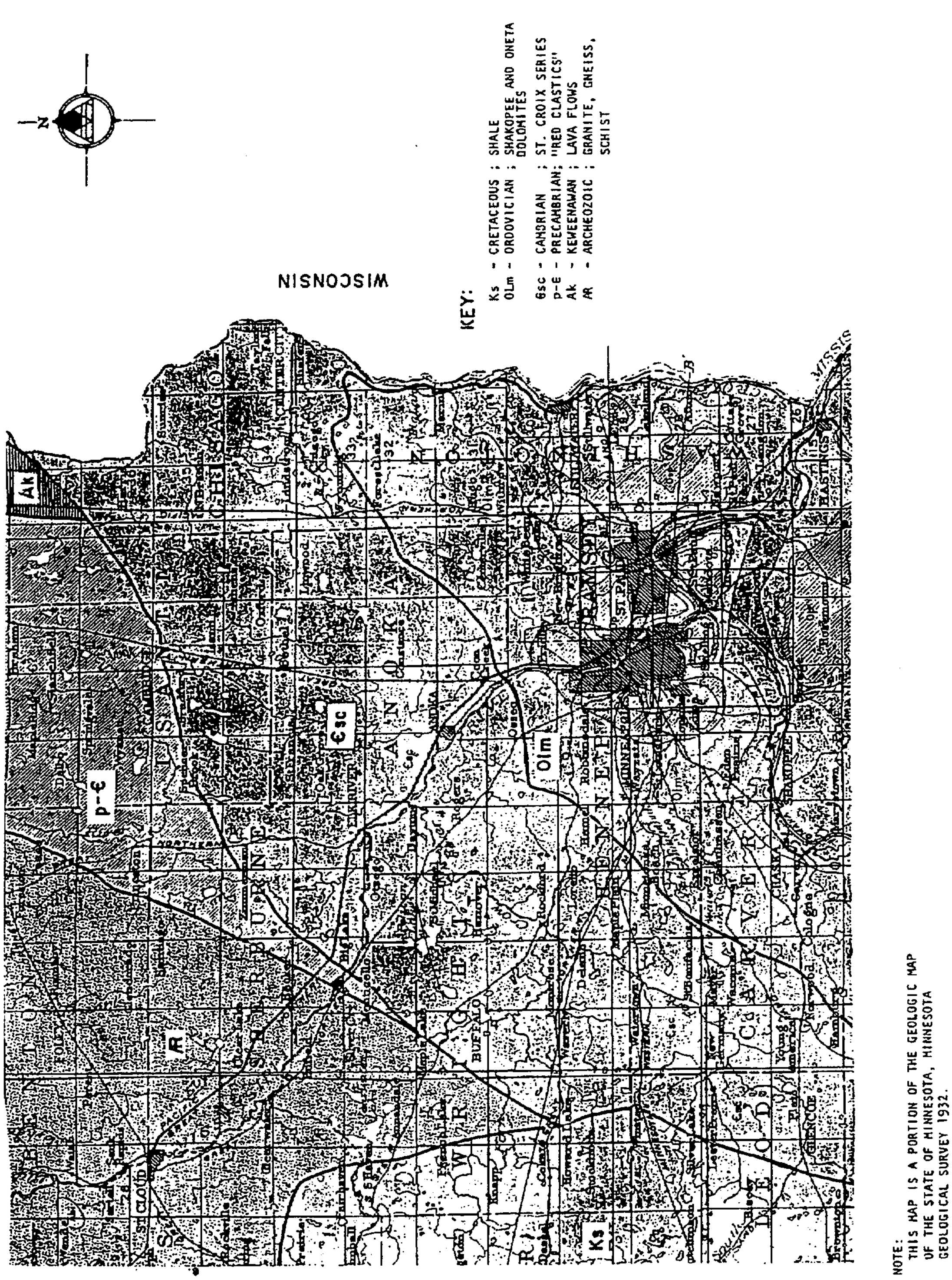
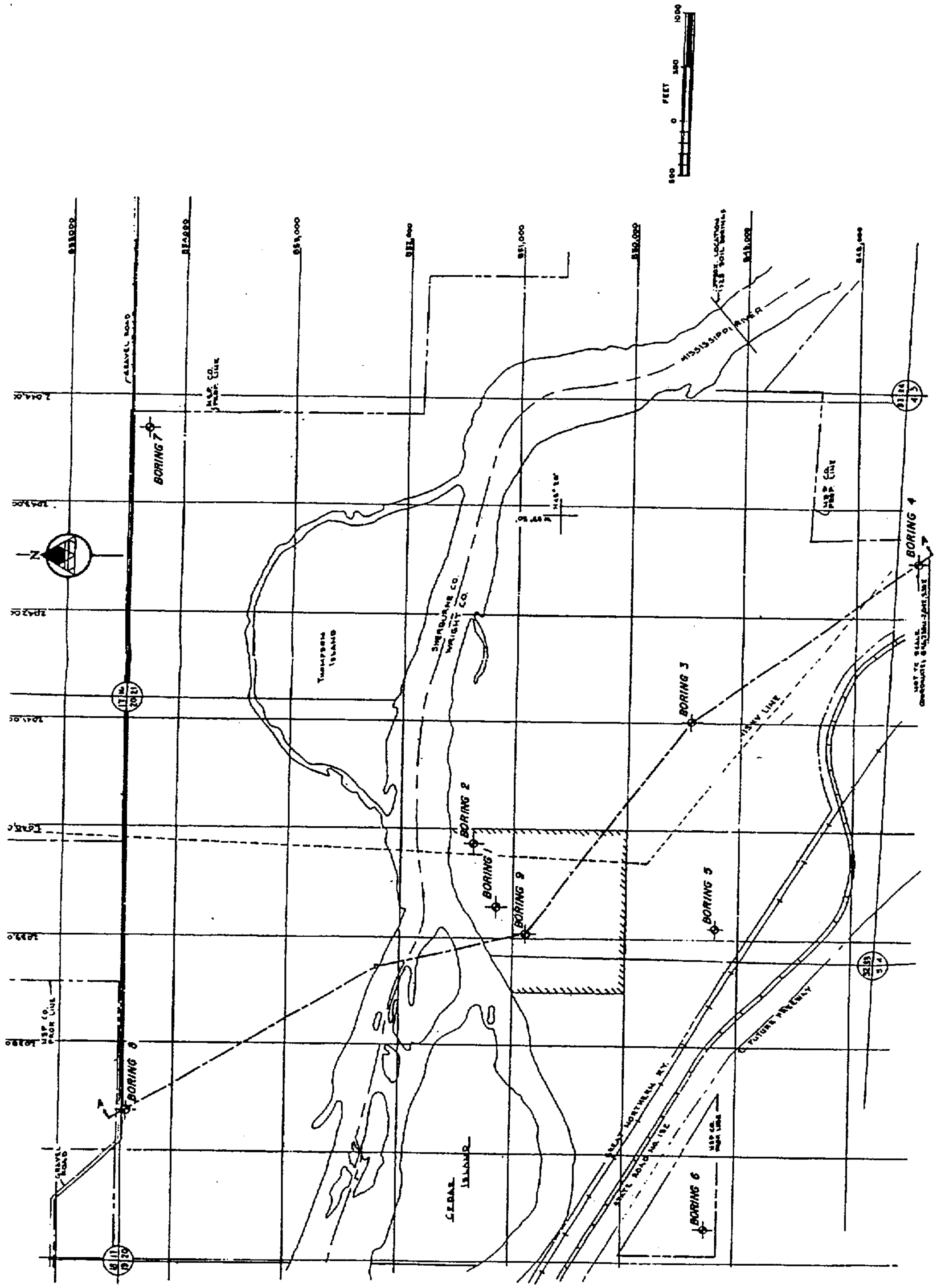


Figure 2.5-2 Location of Original Borings



l/jmr

Figure 2.5-3 Geologic Cross Section A-A

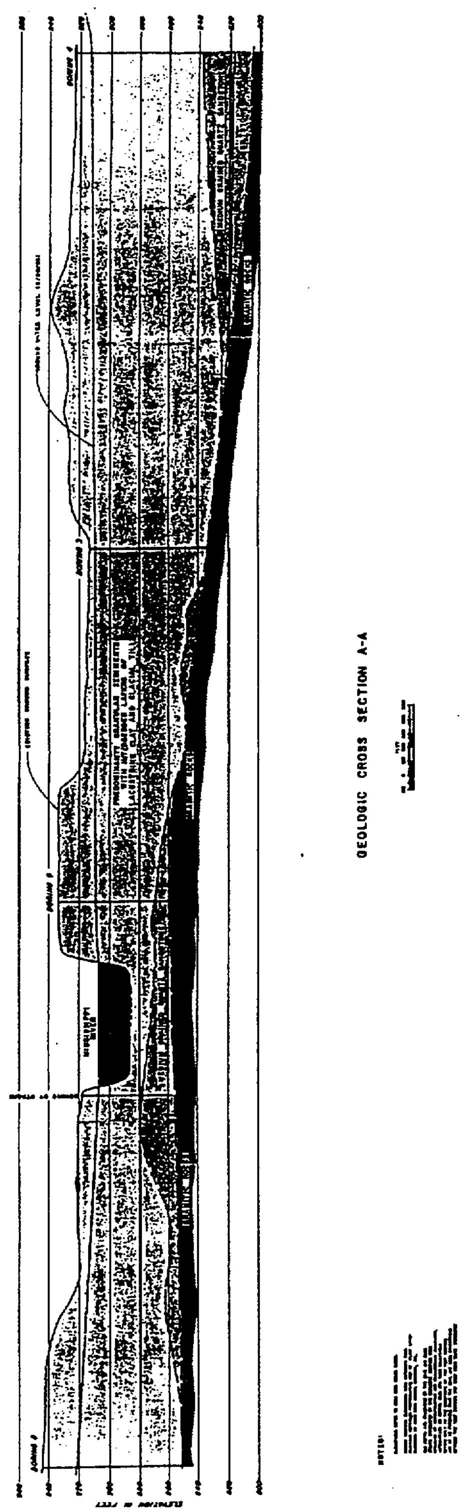
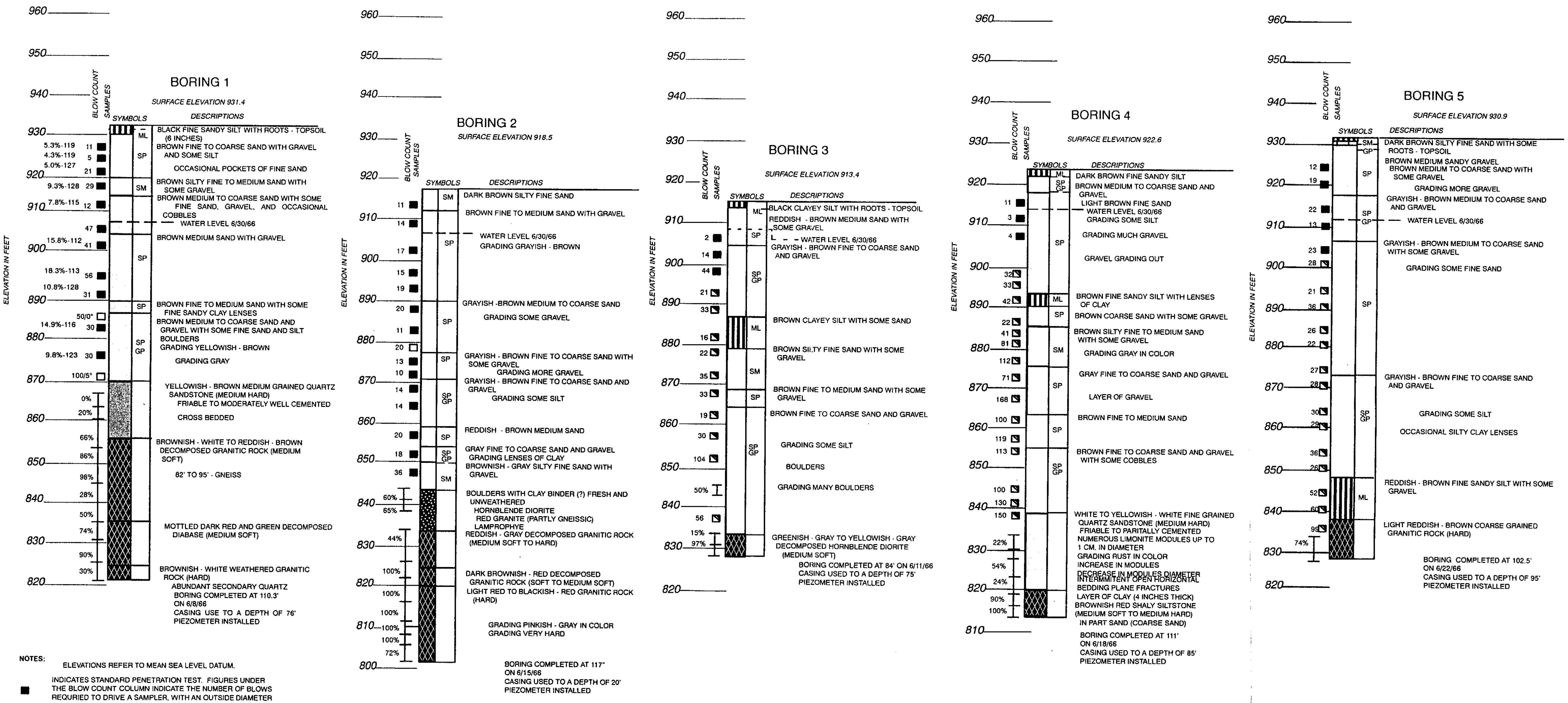


Figure 2.5-4 Log of Borings Sheet 1



NOTES:

ELEVATIONS REFER TO MEAN SEA LEVEL DATUM.

■ INDICATES STANDARD PENETRATION TEST. FIGURES UNDER THE BLOW COUNT COLUMN INDICATE THE NUMBER OF BLOWS REQURIED TO DRIVE A SAMPLER, WITH AN OUTSIDE DIAMETER OF 2 INCHES, 1 FOOT WITH A 140 POUND WEIGHT FALLING TO 30 INCHES.

■ INDATES DAMES & MOORE UNDISTURBED SAMPLE. FIGURES UNDER THE BLOW COUNT COLUMN INDICATE THE NUMBER OF BLOWS REQUIRED TO DRIVE A DAMES & MOORE SAMPLER, WITH AN OUTSIDE DIAMETER OF 3.25 INCHES. 1 FOOT USING A 340 TO 380 POUND WEIGHT FALLING 30 INCHES.

100% INDICATES DEPTH, LENGTH AND PERCENT OF CORE RUN RECOVERED.

THE SOIL DESCRIPTIONS PRESENTED REFER TO THE UNIFIED SOIL CLASSIFICATION SYSTEM.

20.4%-104
PER CENT MOISTURE FIELD EXPRESSED AS A PERCENTAGE OF THE DRY WEIGHT OF SOIL
DRY DENSITY EXPRESSED IN POUNDS PER CUBIC FOOT

Figure 2.5-5 Log of Borings Sheet 2

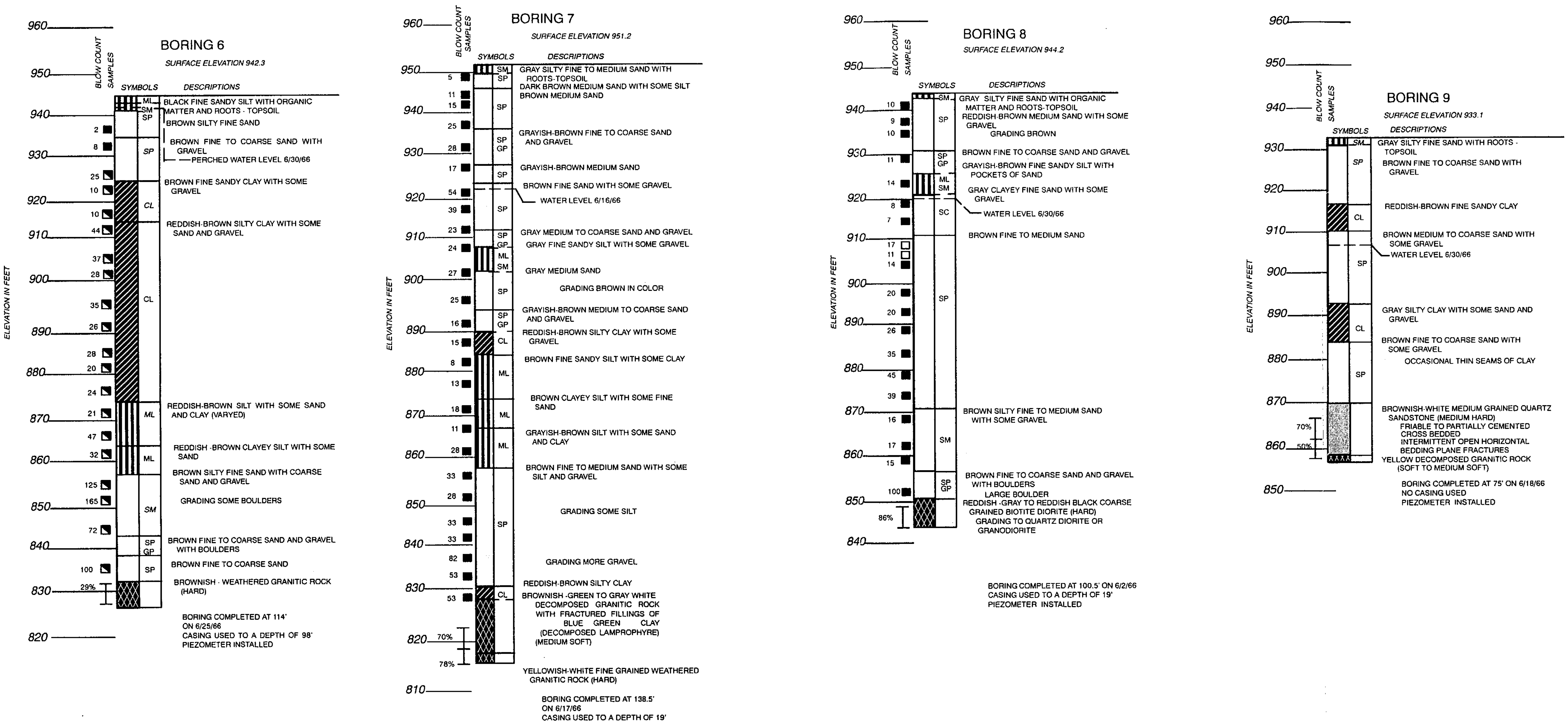


Figure 2.6-1 Principal Earthquakes - Minnesota Region

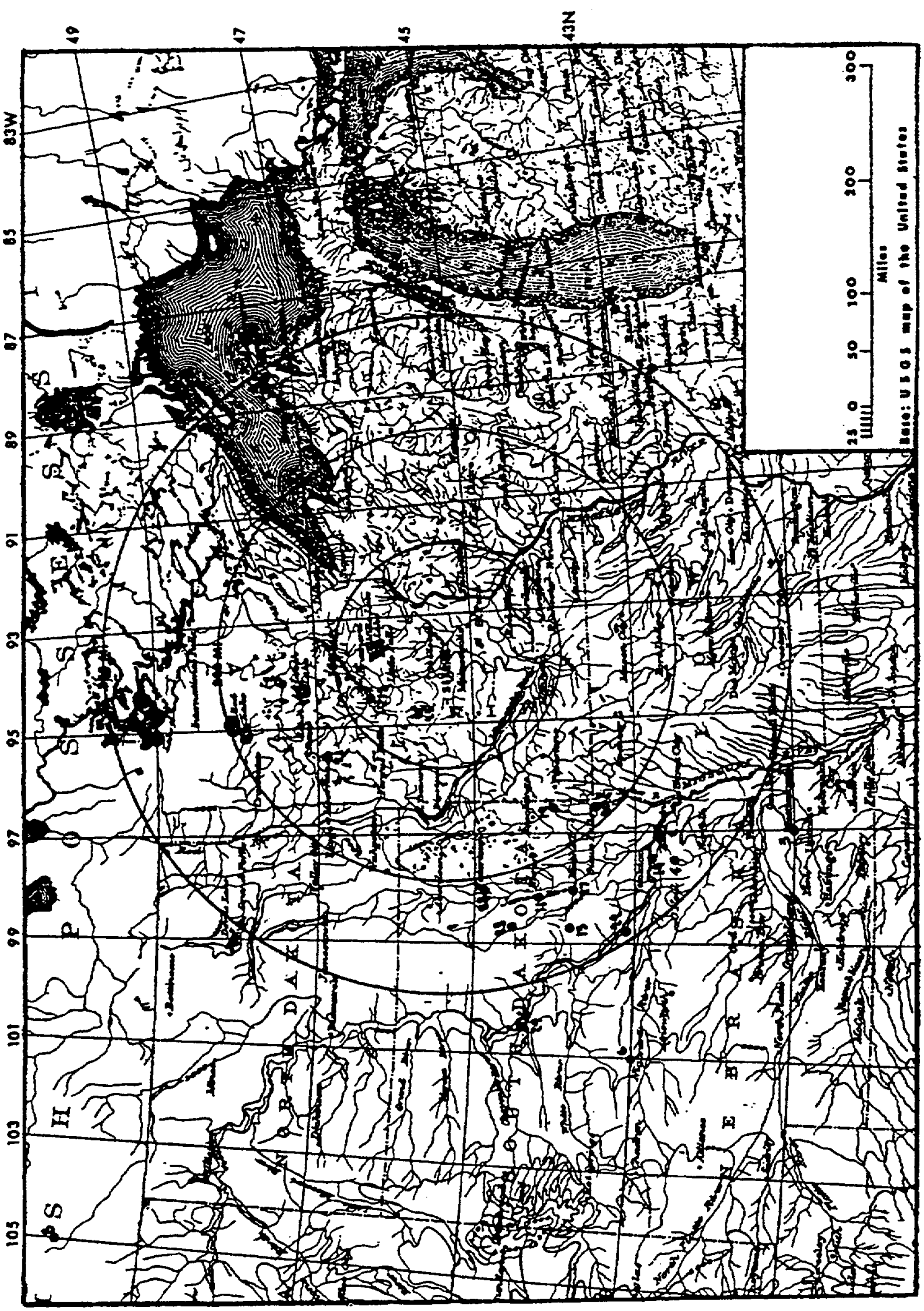


Figure 2.6-2 Tectonic Map of Minnesota Region

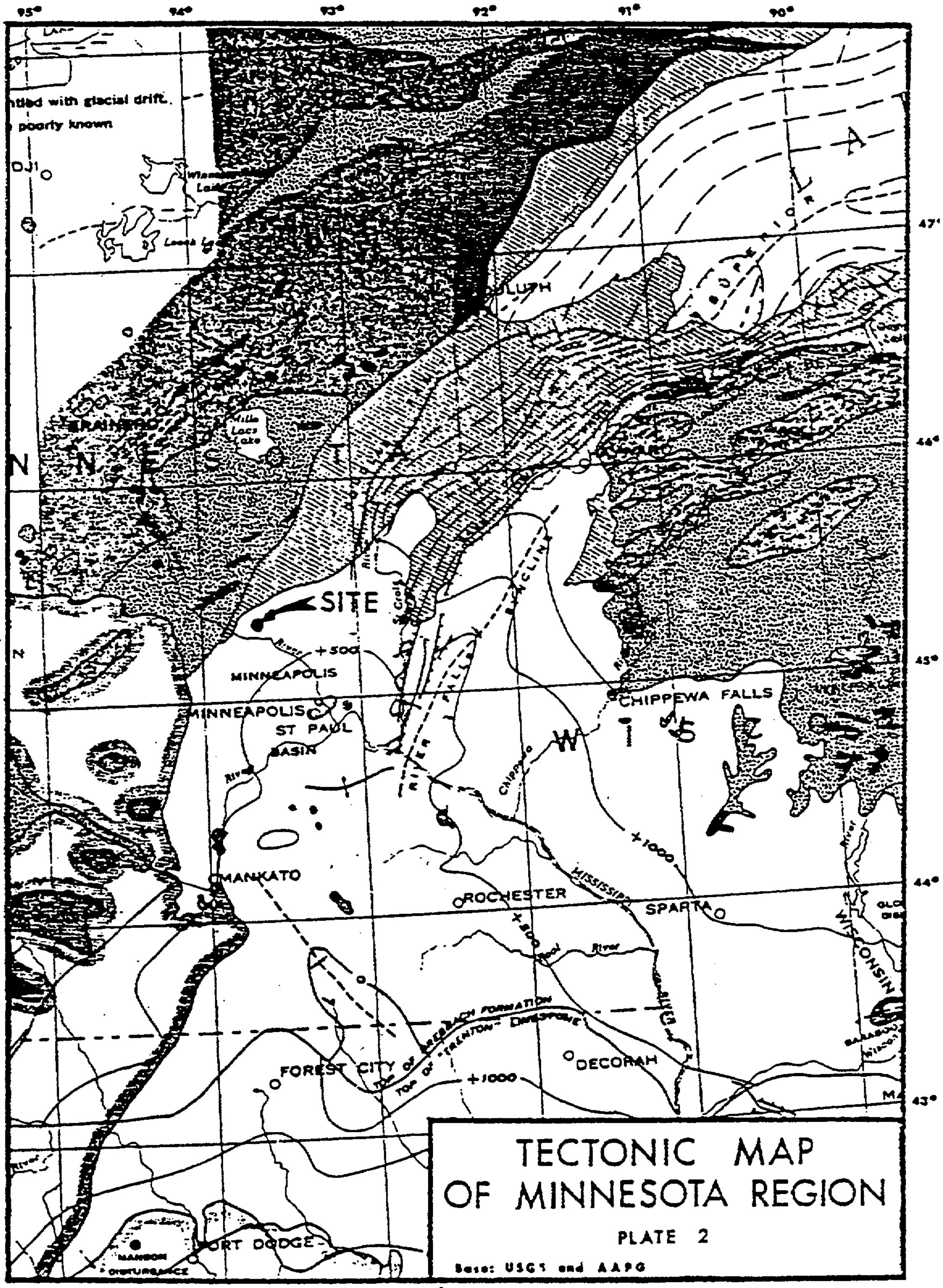


Figure 2.6-3 Seismic Regionalization U.S.A.

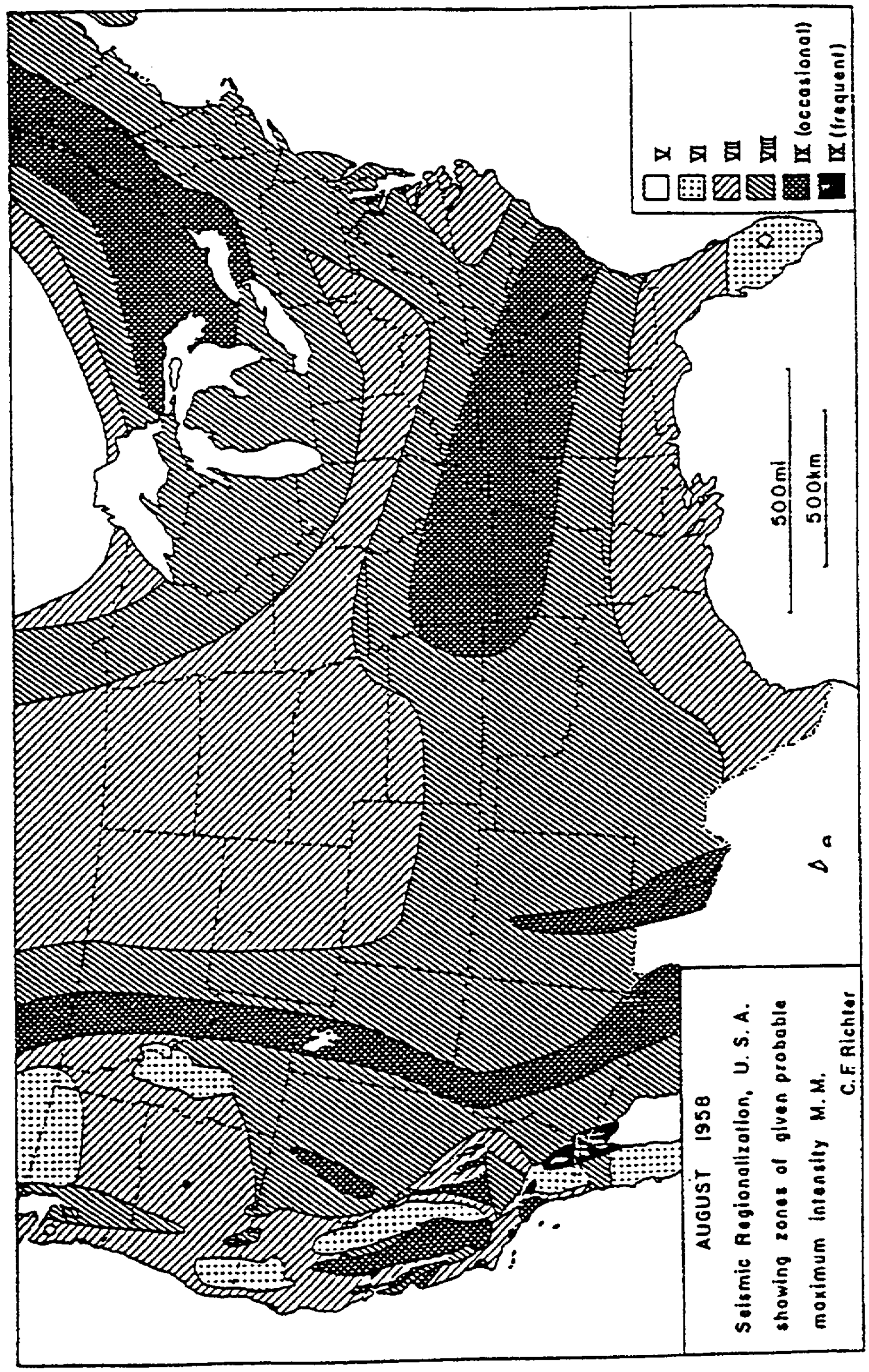


Figure 2.6-4 Seismic Probability Map of U.S.A.

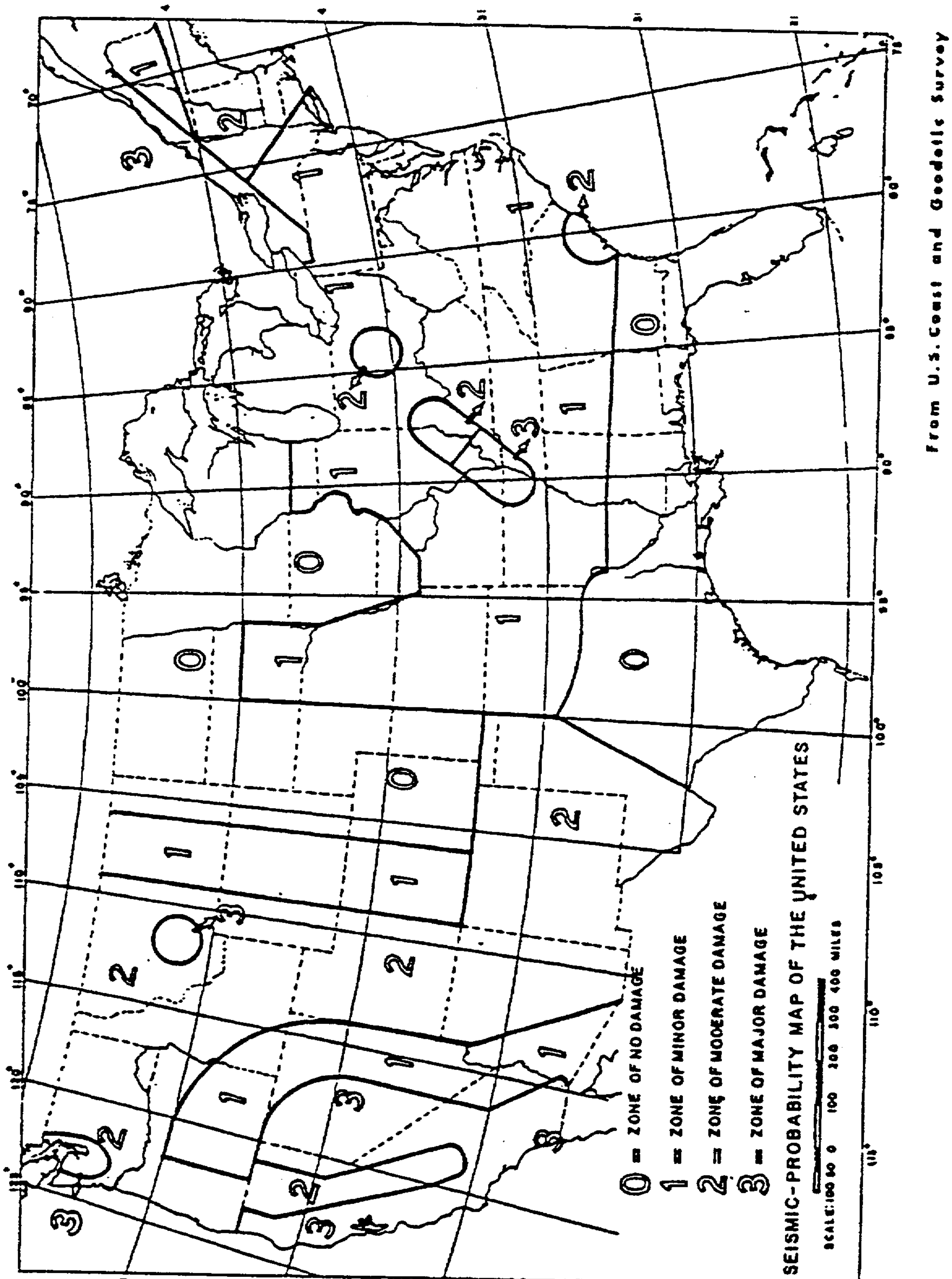


Figure 2.6-5 Seismic Response Spectra