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ENCLOSURES:
Request for a Change to Tech Specs, Appendix A, notarized 11-20-72, for the Monticello Nuclear Generating Plant.

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

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November 20, 1972

Mr. A Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D C 20545

Dear Mr. Giambusso:

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263 LICENSE NO. DPR-22

CHANGE REQUEST DATED NOVEMBER 20, 1972

Attached are three signed originals and 37 conformed copies of a request for a change of Technical Specifications of Appendix A of the Monticello Nuclear Generating Plant. This change request has been reviewed by the Monticello Operations Committee and Safety Audit Committee.

A regular submittal of this type normally requires only three signed originals and 19 conformed copies per 10CFR50.59(d). This request for change includes, in part, the proposed revisions to the reporting requirements section of the Technical Specifications requested in Mr. Donald J Skovholt's letter of September 28, 1972 to Mr. A V Dienhart. The 37 copies attached are responsive to the instructions contained in that letter.

Yours very truly,

L. O. Mayer

L O Mayer, P.E.
Director of Nuclear Support Services

LOM/br

cc: B H Grier
D J Skovholt

Attachments

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UNITED STATES ATOMIC ENERGY COMMISSION

NORTHERN STATES POWER COMPANY

Monticello Nuclear Generating Plant

Docket No. 50-263

REQUEST FOR AUTHORIZATION OF
A CHANGE IN TECHNICAL SPECIFICATIONS
OF APPENDIX A

PROVISIONAL OPERATING LICENSE NO. DPR-22

(Change Request Dated November 20, 1972)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A, Exhibit B, and Exhibit C. Exhibit A describes the proposed changes along with reasons for change. Exhibit B consists of copies of pages of the Technical Specifications marked up to indicate a portion of the proposed changes. Exhibit C consists of newly prepared pages of the Technical Specifications which present the rest of the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Arthur V. Dienhart
Arthur V Dienhart
Vice President-Engineering

On this 20 day of November, 1972 before me a notary public in and for said County, personally appeared Arthur V. Dienhart, Vice President - Engineering, and being first duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he has read it and knows the contents thereof, that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

John J. Smith
John J Smith
Notary Public, Hennepin County, Minnesota

JOHN J. SMITH
Notary Public, Hennepin County, Minnesota
My Commission Expires March 3, 1976

UNITED STATES ATOMIC ENERGY COMMISSION

NORTHERN STATES POWER COMPANY

Monticello Nuclear Generating Plant Docket No. 50-263

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A CHANGE IN TECHNICAL SPECIFICATIONS
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NORTHERN STATES POWER COMPANY

By /s/ Arthur V Dienhart
Arthur V Dienhart
Vice President-Engineering

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/s/ John J Smith
John J Smith
Notary Public, Hennepin County, Minnesota

NOTARY SEAL

My Commission Expires March 3, 1976

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

CHANGE REQUEST DATED NOVEMBER 20, 1972
PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF PROVISIONAL OPERATING
LICENSE NO. DPR-22

Pursuant to 10 CFR 50.59, the holders of the above-mentioned license hereby propose the following changes to Appendix A, Technical Specifications:

Proposed Change

On page 81, Section 3.3.E, change ".... Specification 6.6." to read ".... Specification 6.7.B."

On page 90, Section 3.4.B.2, change ".... Specification 6.6.E, ..." to read ".... Specification 6.7.C.2, ..."

On page 131, Bases 3.6 and 4.6, change ".... Specification 6.6.E.3.d." to read ".... Specification 6.7.C.1.d."

On page 136, Bases 3.6 and 4.6, change ".... Specification 6.6.E.3." to read ".... Specification 6.7.C.1.c."

Reason for Change

Substantial revisions to Section 6.0 of the Technical Specifications are proposed later in this change request. The above revisions are necessary so that the references conform to the revised paragraph numbering in Section 6.0.

Proposed Change

On page 108, Section 3.5.H, delete the whole paragraph "When it is determined .... out-of-service period." and replace it with the following: "When it is determined that maintenance to restore components or systems to an operable condition will last longer than the periods specified, the reporting requirements of 6.7.C.2 shall be fulfilled."

Reason for Change

Current paragraphs 3.5.H and 6.6.E appear to require two separate reports on the same subject (i.e., extended maintenance of certain components and systems). "The determination" as referenced in each of these paragraphs probably will not be made until some portion of the allowable out-of-service period has elapsed. The due date for one report is prior to the end of the allowable out-of-service period (seven days in most cases) and the other is due within seven days of "the determination." This inconsistency or apparent dual reporting requirement plus a desire to collect all reporting requirements and their definitions in Section 6.0 is the reason for requesting this change. The proposed wording for this reporting requirement is contained in Section 6.7.C.2 of this change request.

Proposed Change

Page 1, Sections 1.0.A, 1.0.B, and 1.0.C, is to be replaced by pages 1 and 1A, Sections 1.0.A, 1.0.B, and 1.0.C, as contained in Exhibit C.

Pages 192 - 211, Section 6.0, Administrative Controls, are to be replaced by Pages 192 - 220 as contained in Exhibit C.

Reason for Change

The Technical Specifications, Appendix A of Provisional Operating License No. DPR-22, have been in effect for over two years. Certain portions of Section 6.0, Administrative Controls, have been difficult to interpret, particularly Table 6.1.1 which tends to put the review committees in a line function rather than in a review and audit function. Our operating and reporting experience under Section 6.0 of the Technical Specifications has also identified portions which could be refined for clearer interpretation and improved workability. Certain changes in the structures of the organizations, shown in Figure 6.6.1 and 6.6.2, were implemented in 1972 as reported in the NSP letter of March 10, 1972 to Dr. Peter A Morris. During the past year, Directorate of Licensing personnel have expressed a desire for certain changes to this section of the Technical Specifications. A letter dated September 28, 1972 from Mr. D J Skovholt of AEC-DL to Mr. A V Dienhart of NSP requested that we submit, within sixty days, proposed revisions to the reporting requirements in the Administrative Controls of our Technical Specifications which meet the guidance set forth in the enclosures to the subject letter.

NSP representatives have participated in a number of discussions with AEC-Licensing and AEC-Regulatory Operations personnel in regard to a more acceptable and somewhat "standardized" format for this section of the Technical Specifications. Based on these discussions and on in-depth guidance from the ANS Proposed Standard for Administrative Controls for Nuclear Plants, Section 1.0, Definition of Abnormal Occurrences, and Section 6.0, Administrative Controls, have been rewritten as contained in Exhibit C attached.

The September 28, 1972 letter from Mr D J Skovholt on reporting requirements requested that additional information be submitted on those Technical Specification revisions and facility measurements for which it is not practical to follow the guidance set forth in that letter. We believe that Section 6.7, Plant Reporting Requirements, in this change request meets the intent of the guidance provided.

The guidance given for Section 6.7.A.2.g on environmental monitoring reporting in the semi-annual operating report is more extensive than that required in our current Technical Specification. Environmental monitoring sampling can continue right up to the end of the six month period. Sample analysis, tabulation and statistical analysis of the data, evaluation of the data, and preparation of a final draft of the environmental monitoring section of the semi-annual report in the detail proposed within the 60 day time period allotted appear difficult, if not impracticable. We believe that the annual Environmental Monitoring Report is the more appropriate location for this type of statistical analysis and evaluation of its significance, particularly since it treats a whole year with a full cycle of seasonal variations as opposed to the environmentally non-related six month operating report periods. If the semiannual operating report were to report environmental monitoring data only, without detailed analysis, this could be practicably accomplished in the 60 day time period. In spite of this apparent difficulty, we have used the guidance provided by the September 28, 1972 letter for Section 6.7.A.2.g of this change request and will evaluate the extent of the burden imposed following the first applicable reporting period.

EXHIBIT B

(5 pages)

3.0 LIMITING CONDITIONS FOR OPERATION

E. Reactivity Anomalies

At a specific steady state base condition of the reactor actual control rod inventory will be periodically compared to a normalized computed prediction of the inventory. If the difference exceeds one percent, Δk, reactor power operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The AEC shall be notified within 24 hours of this situation in accordance with Specification ~~6.6.~~ 6.7.B.

F. If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

E. Reactivity Anomalies

During the startup test program and at each startup following refueling outages, the actual rod inventory shall be compared to a normalized computed prediction of the inventory. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the actual rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

3.0 LIMITING CONDITIONS FOR OPERATION

inoperable, Specification 3.4.A shall be considered fulfilled, provided that:

1. The component is returned to an operable condition within 7 days, or
2. A written report, as specified in Specification ~~6.6.E,~~ 6.7.C.2, shall be submitted to the Atomic Energy Commission when the maintenance to restore the component to an operable condition will last longer than 7 days.

C. Volume-Concentration Requirements

The liquid poison tank shall contain a boron bearing solution that satisfies the volume-concentration requirements of Figure 3.4.1 and at all times when the standby liquid poison system is required to be operable the temperature shall not be less than the solution temperature presented in Figure 3.4.2. In addition, the heat tracing on the pump suction lines shall be operable whenever the room temperature is less than the solution temperature presented in Figure 3.4.2.

4.0 SURVEILLANCE REQUIREMENTS

C. The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. <u>At least once per month</u> -

   Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. When irradiated fuel is in the reactor vessel and reactor coolant temperature is less than 212^{O}F, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel. H. Extended Maintenance When it is determined that maintenance to restore components or systems to an operable condition will last longer than the periods specified, ~~a report detailing the circumstances and the estimated date for returning the components or systems to an operable condition shall be submitted to the AEC prior to the allowable end of the out-of-service period.~~ the reporting requirements of 6.7.C.2 shall be fulfilled.	

Bases 3.6 and 4.6 - Continued:

The initial NDT Temperature of the vessel shell material opposite the reactor core region is $0^{\circ}F$. The initial NDT temperature in the main dome flanges, and the shell and head material connecting to these flanges is $10^{\circ}F$, and elsewhere is $40^{\circ}F$. The design life of the reactor vessel is 40 years and the maximum fast neutron exposure at 40 years is calculated to be 5.4×10^{17} nvt. The NDT temperature limit curve in Figure 4.6.1 uses the "worst case" curve of the FSAR to establish the NDT temperature shift and is therefore conservative. The expected NDT temperature shift for this vessel at 5.4×10^{17} nvt is expected to be $0^{\circ}F$. Figure 4.6.1 also incorporates a $60^{\circ}F$ factor of safety. This factor is based upon the requirements of the ASME code and the considerations which resulted in these requirements. Therefore, the specification provides for "worst case" data as well as $60^{\circ}F$ of margin to provide assurance that operation in the non-ductile region will not occur.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak-tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. Both the head and the headflange have an NDT temperature of $10^{\circ}F$, and they are not subject to any appreciable neutron radiation exposure. Therefore, the minimum vessel head and head flange temperature for bolting the head flange and vessel flange is established as $10^{\circ}F + 60^{\circ}F$ or $70^{\circ}F$.

Numerous data are available relating integrated flux and the change in nil-ductility transition temperature (NDTT) in various steels. The most conservative data has been used in Specification 3.3. The integrated flux at the vessel wall is calculated from core physics data and will be measured using flux monitors installed inside the vessel. The measurements of the neutron flux at the vessel wall will be used to check and if necessary correct, the calculated data to determine an accurate NDTT.

In addition, vessel material samples will be located within the vessel to monitor the affect of neutron exposure on these materials. The samples include specimens of base metal, weld zone metal, heat affected zone metal, and standard specimens. These samples will receive neutron exposure more rapidly than the vessel wall material and therefore will lead the vessel in integrated neutron flux exposure. These samples will provide further assurance that the shift in NDTT used in the specification is conservative. An analysis and report will be submitted to the AEC on all such surveillance specimens removed from the reactor vessels in accordance with Specification ~~6.6.E.3.d.~~ These reports shall include the information specified in ASTM E-185-66, "Recommended Practices for Surveillance Tests on Structural Materials in Nuclear Reactor," and information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.

6.7.C.1.d.

Bases Continued 3.6 and 4.6:

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

The program outlined in Table 4.6.1 is limited to inspections of the primary coolant system. It is anticipated that the data collected during the first five years of operation will provide a suitable basis to evaluate the need for inspecting other portions of the facility (such as the main steam lines downstream of the main steamline isolation valves). These data along with the overall operating experiences will be reviewed to determine the inspection program to be implemented for the lifetime of the facility. The results of this study together with the proposed lifetime inspection program will be submitted to the AEC in accordance with Specification ~~6.6.E.3.~~ 6.7.C.1.c.

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

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

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the AEC. These studies show that it requires thousands of stress cycles at

EXHIBIT C

Sections 1.0.A, 1.0.B and 1.0.C consisting of pages 1 and 1A.

Section 6.0 consisting of pages 192 through 220 (29 pages).

INTRODUCTION

These Technical Specifications are prepared in accordance with the requirements of 10CFR50.36 and apply to the Monticello Nuclear Generating Plant, Unit No. 1. The bases for these Specifications are included for information and understandability purposes.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the Specifications may be achieved.

A. Abnormal Occurrence - An abnormal occurrence means the occurrence of any plant condition that results in:

1. A safety system setting less conservative than the limiting setting established in the Technical Specifications.
2. Violation of a limiting condition for operation established in the Technical Specifications.
3. An uncontrolled or unplanned release of radioactive material from any plant system designed to act as a boundary for such material in an amount of significance with respect to limits prescribed in Technical Specifications.
4. Failure of one or more components of an engineered safety feature or plant protection system that causes or threatens to cause the feature or system to be incapable of performing its intended function.
5. Abnormal degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process.
6. Uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta k/k$.
7. Observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the plant.

8. Conditions arising from natural or offsite man-made events that affect or threaten to affect the safe operation of the plant.

B. <u>Alteration of the Reactor Core</u> - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. (Normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations.)

C. <u>Hot Standby</u> - Hot Standby means operation with the reactor critical in the startup mode at a power level just sufficient to maintain reactor pressure and temperature.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

A. The Plant Manager has the overall full-time onsite responsibility for safe operation of the facility. During periods when the Plant Manager is unavailable, he may delegate this responsibility to other qualified supervisory personnel.

B. The Northern States Power corporate organizational structure relating to the operation of this plant is shown in Figure 6.1.1.

C. The minimum functional organization for operation of the plant shall be as shown in Figure 6.1.2. The minimum shift complement for special plant conditions shall be as follows:

1. The unit shutdown and the reactor contains fuel: a licensed senior reactor operator on site; a licensed reactor operator in the control room; and a plant operator on site.

2. During cold startup, while shutting down the reactor and during recovery from any reactor trip: a minimum of two licensed operators in the control room.

3. Fuel handling operations: a licensed senior reactor operator shall be directly in charge of any fuel handling operations and shall have no other concurrent duties.

4. Any fuel on site: an individual who meets the qualification of a health physics technician shall be on site at all times that there is fuel on site.

D. Minimum qualifications, training, replacement training and retraining of plant personnel shall be in accordance with that stated in the Standard for Selection and Training of Personnel for Nuclear Power Plants, ANSI N18.1 - 1971. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

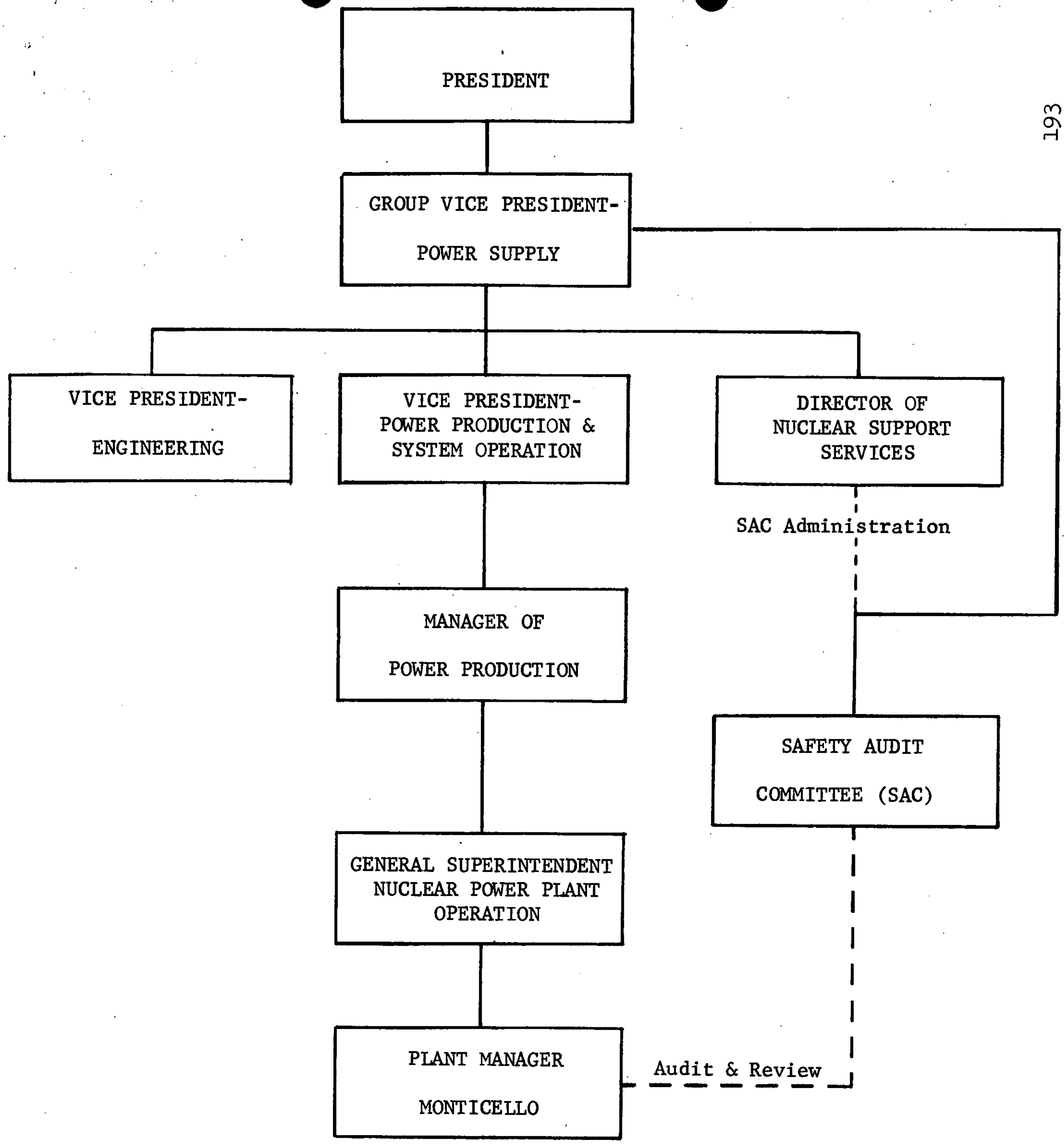
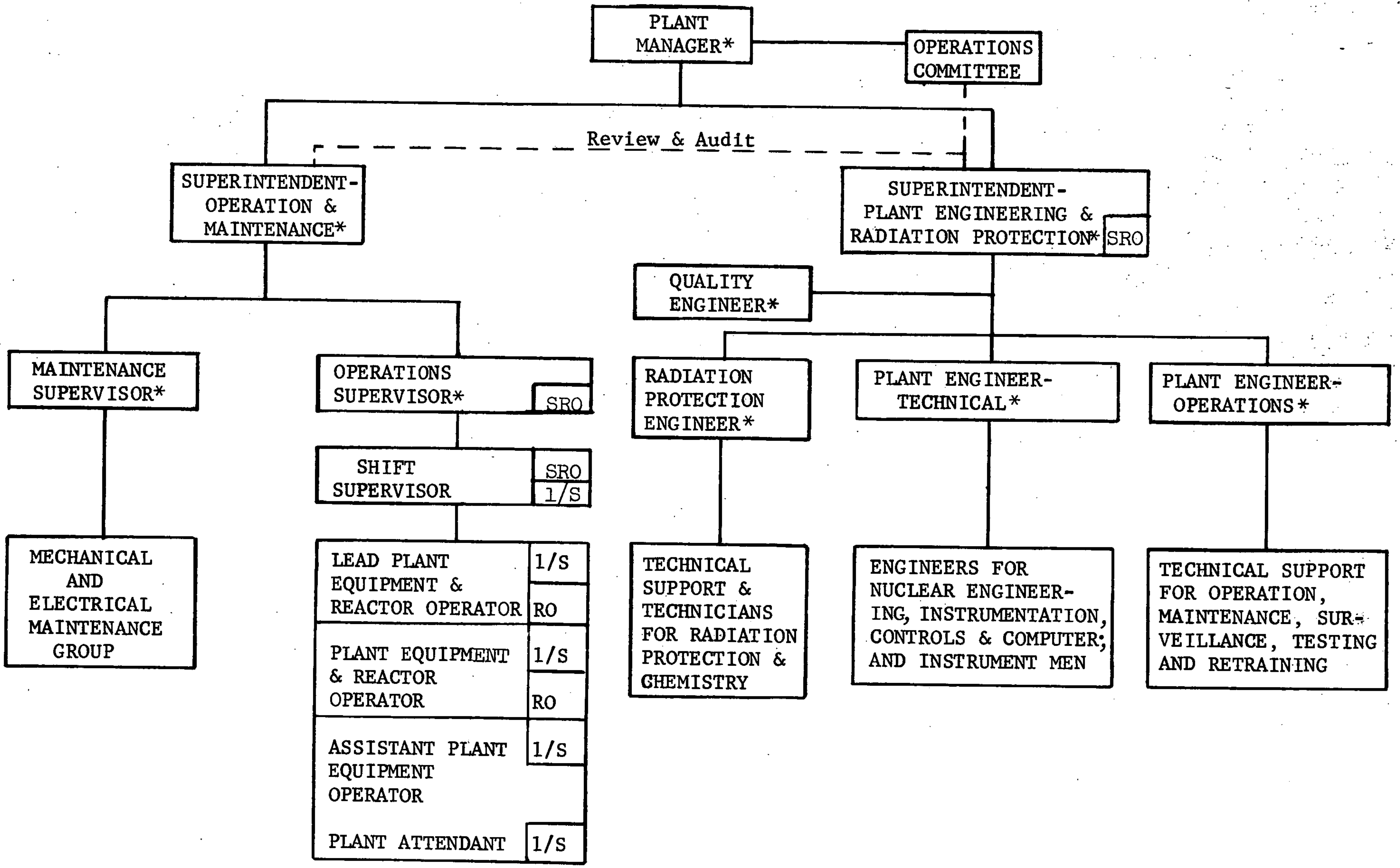


FIGURE 6.1.1

NSP CORPORATE ORGANIZATIONAL

RELATIONSHIP TO ON-SITE OPERATING ORGANIZATION



CODE:

*	-	Key Supervisor
1/S	-	one/shift
RO	-	Licensed Reactor Operator
SRO	-	Licensed Senior Reactor Operator

MONTICELLO NUCLEAR GENERATING PLANT
Minimum Functional Organization For
On-Site Operating Group

FIGURE 6.1.2

6.2 Review and Audit

Organizational units for the review and audit of facility operations shall be constituted and have the responsibilities and authorities outlined below:

A. Safety Audit Committee (SAC)

The SAC must: verify that operation of the plant is consistent with company policy and rules, approved operating procedures and operating license provisions, review important proposed plant changes, tests and procedures; verify that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

Audits of selected aspects of plant operation shall be performed with a frequency commensurate with their safety significance and in such a manner as to assure that an audit of all nuclear safety related activities is completed within a period of two years. Periodic review of the audit programs should be performed by the SAC at least twice a year to assure that such audits are being accomplished in accordance with requirements of Technical Specifications. The audits shall be performed in accordance with appropriate written instructions or procedures and should include verification of compliance with internal rules, procedures (for example, normal, off-normal, emergency, operating, maintenance, surveillance, test and radiation control procedures and the emergency and security plans), regulations involving nuclear safety and operating license provisions; training, qualification and performance of operating staff; and corrective actions following abnormal occurrences or unusual events. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. Written reports of such audits shall be reviewed at a schedule meeting of the SAC and by appropriate members of management including those having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, shall be taken when indicated.

1. Membership

   a. The SAC shall consist of at least five (5) persons.

b. The SAC Chairman shall be a NSP management representative appointed by the Group Vice President-Power Supply. Other SAC members shall be appointed by the Group Vice President-Power Supply or such other person as he may designate. The Chairman shall appoint a Vice Chairman from the SAC membership to act in his absence.

c. No more than two members of the SAC shall be from groups holding line responsibility for operation of the plant.

d. The SAC members should collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. When the nature of a particular problem dictates, special consultants will be utilized, as necessary, to provide expert advice to the SAC.

2. Meeting Frequency

The SAC shall meet on call by the Chairman but not less frequently than twice a year.

3. Quorum

A majority of the permanent members, including the Chairman or Vice Chairman. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Responsibilities - The following subjects should be reported to and reviewed by the SAC:

a. Proposed tests and experiments, and their results, when such tests or experiments may constitute an unreviewed safety question, as defined in Section 50.59, Part 50, Title 10, Code of Federal Regulations.

b. Evaluations of proposed changes to procedures, equipment and systems completed under provisions of Section 50.59, Part 50, Title 10, Code of Federal Regulations to verify that such proposed changes do not constitute unreviewed safety questions.

c. Proposed changes in Technical Specifications or operating license.

d. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, operating license requirements, or of internal procedures or instructions, having safety significance.

e. Significant operating abnormalities or deviations from normal and expected performance of plant equipment.

f. Abnormal occurrences as defined in these Technical Specifications.

g. Receipt of information indicating that there may be an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

h. Operations Committee proceedings and minutes to determine if matters considered by that Committee involve unreviewed or unresolved safety questions.

i. Training, qualification and performance of operating staff.

j. Disagreement between the recommendations of the Operations Committee and the Plant Manager.

k. Security and emergency plans and their implementing procedures.

l. Environmental Monitoring Program and its results

m. Quality Assurance program and evaluate its adequacy.

Review of events covered under 4.d-4.g above shall include reporting to appropriate members of management on the results of investigations and recommendations to prevent or reduce the probability of recurrence.

5. Authority

The SAC shall be advisory to the Group Vice President-Power Supply.

6. Records

Minutes shall be prepared and retained for all scheduled meetings of the Safety Audit Committee and shall identify all documentary material reviewed. There shall be a timely dissemination of minutes to the Group Vice President-Power Supply, the General Superintendent of Nuclear Power Plant Operation, each member of the SAC and others designated by the Chairman or Vice Chairman.

7. Procedures

A written charter for the SAC shall be prepared that contains:

a. Subjects within the purview of the group

b. Responsibility and authority of the group including responsibility to identify problems and to recommend solutions to appropriate management

c. Mechanisms for convening meetings

d. Provisions for any use of subgroups

e. Authority to obtain access to the nuclear power plant operating record files and operating personnel to perform the audit function

f. Requirements for distribution of reports and minutes prepared by the group to others in the NSP organization

g. Identification of the management position to which the SAC reports

h. Provisions for assuring that the SAC is kept informed on a timely basis of matters within its purview

i. Provisions for a formal approval of the minutes

B. Operations Committee (OC)

1. Membership

The Operations Committee shall consist of at least six (6) members drawn from the on-site supervisory staff. The Plant Manager shall serve as Chairman of the OC and shall appoint a Vice Chairman from the OC membership to act in his absence.

2. Meeting Frequency

The Operations Committee will meet on call by the Chairman or as requested by individual members and at least monthly.

3. Quorum

A majority of the permanent members, including the Chairman or Vice Chairman.

4. Responsibilities - The following subjects shall be reviewed by the Operations Committee:

a. Proposed tests and experiments and their results

b. Modifications to plant systems or equipment as described in Final Safety Analysis Report and having nuclear safety significance.

c. Proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that are determined by the Plant Manager to affect nuclear safety.

d. Proposed changes to the Technical Specifications or operating license.

e. All reported or suspected violations of Technical Specifications, operating license requirements, administrative procedures, operating procedures. Results of investigations, including evaluation and recommendations to prevent recurrence, will be reported, in writing, to the General Superintendent of Nuclear Power Plant Operation and to the Chairman of the Safety Audit Committee.

f. Abnormal occurrences as defined in these Technical Specifications and unusual events.

g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with off-site support groups.

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan, and the Security Plan, shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.

i. Perform special reviews and investigations, as requested by the Safety Audit Committee.

5. Authority

The OC shall be advisory to the Plant Manager. In the event of disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the General Superintendent of Nuclear Power Plant Operation and the Chairman of the SAC for review.

6. Records

Minutes shall be recorded for all meetings of the OC. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Superintendent of Nuclear Power Plant Operation and others designated by the OC Chairman or Vice Chairman.

7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group

b. Content and method of submission of presentations to the Operations Committee

c. Mechanism for scheduling meetings

d. Meeting agenda

e. Use of subcommittees

f. Review and approval, by members, of OC actions

g. Distribution of minutes

6.3 Actions to be Taken in the Event of An Abnormal Occurrence in Plant Operation

A. Any abnormal occurrence as defined in these Technical Specifications shall be reported to the General Superintendent of Nuclear Power Plant Operation, or his designated alternate in his absence, and shall be reviewed by the Operations Committee. A separate, sequentially numbered, report shall be prepared for each abnormal occurrence. Each report shall describe the circumstances leading up to and resulting from the occurrence, the corrective action taken, an attempt to define the cause of the occurrence and shall recommend appropriate action to prevent or reduce the probability of a repetition of the occurrence. The Operations Committee shall review the report and recommend further action, if necessary.

Copies of all such reports and Operations Committee recommendations shall be submitted to the Chairman of the Safety Audit Committee for review and to the General Superintendent of Nuclear Power Plant Operation for review and approval of any recommendations. All abnormal occurrences shall be reported as specified in Section 6.7 - Reporting Requirements.

6.4 Action to be Taken if a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down immediately. An immediate report shall be made to the General Superintendent of Nuclear Power Plant Operation, or his designated alternate in his absence, and to the Director of the Region III Regulatory Operations Office. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations by the Operations Committee, shall also be prepared. This report shall be submitted to the General Superintendent of Nuclear Power Plant Operation and the Chairman of the Safety Audit Committee.

Reactor operation shall not be resumed until authorized by the US Atomic Energy Commission.

6.5 Plant Operating Procedures

Detailed written procedures, including the applicable check-off lists and instructions, covering areas listed below shall be prepared and adhered to. These procedures and changes thereto, except as specified in 6.5.D, shall be approved by the Operation Committee.

A. Plant Operations

1. Integrated and system procedures for normal startup, operation and shutdown of the reactor and all systems and components involving nuclear safety of the facility.

2. Fuel handling operations

3. Actions to be taken to correct specific and foreseen potential or actual malfunction of systems or components including responses to alarms, primary system leaks and abnormal reactivity changes and including follow-up actions required after plant protective system actions have initiated.

4. Surveillance and testing requirements that could have an effect on nuclear safety.

5. Implementing procedures of the security plan.

6. Implementing procedures of the emergency plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity.

Drills on the procedures specified in A.3 above shall be conducted as a part of the retraining program. Drills on the procedures specified in A.6 above shall be conducted at least semiannually, including a check of communications with offsite support groups.

B. Radiological

Radiation control procedures shall be maintained and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

1. a. Paragraph 20.203 "Caution signs, labels, signals and controls." In lieu of the "Control device" or alarm signal required by paragraph 20.203 (c) (2), each high radiation area in which the intensity of radiation is 1000 mRem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mRem/hr, except that locked doors shall be provided to prevent unauthorized entry into these areas and the keys to these locked doors shall be maintained under the administrative control of the Plant Manager.

2. Pursuant to 10 CFR 20.103 (c) (1) and (3), allowance can be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

   a. The limits provided in Section 20.103 (a) and (b) are not exceeded.

   b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column 1, of 10 CFR 20.

   c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified is based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in 20.101. These materials shall be subject to applicable process and other engineering controls.

3. In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in paragraph 2 of this section are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment does not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.5.1, appended to this Specification, for the respiratory protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the latter quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

(1) Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures and to permit proper selection of respiratory protective equipment.

(2) Written procedures to assure proper selection, supervision and training of personnel using such protective equipment.

(3) Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

(4) Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair and storage.

(5) Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

(6) Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U.S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.5.1 below. Equipment not approved under U.S. Bureau of Mines Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines approved equipment of the same type, as specified in Table 6.5.1 below.

f. Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table 6.5.1 below in selecting and using respiratory protective equipment.

4. These Specifications with respect to the provisions of 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20 Section 20.103, which would make this Specification unnecessary.

TABLE 6.5.1
PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR § 14.4 (b) (4)
Facepiece, full 7/	NP	100	21B 30 CFR § 14.4 (b) (5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR § 12.2 (c) (2) Type C (i)
Facepiece, full	CF	1,000	19B 30 CFR § 12.2 (c) (2) Type C (i)
Facepiece, full 7/	D	500	19B 30 CFR § 12.2 (c) (2) Type C (ii)
Facepiece, full	PD	1,000	19B 30 CFR § 12.2 (c) (2) Type C (iii)
Hood	CF	5/	6/
Suit	CF	5/	6/
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 7/	D	500	13E 30 CFR § 11.4 (b) (2) (i)
Facepiece, full	PD	1,000	13E 30 CFR § 11.4 (b) (2) (ii)
Facepiece, full	R	1,000	13E 30 CFR § 11.4 (b) (1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR § 12.2 (e) or applicable schedules as listed above

1/, 2/, 3/, 4/, 5/, 6/, 7/, (These notes are on the following pages)

1/ See the following symbols

CF:	Continuous Flow	
D:	Demand	
NP:	Negative Pressure	(i.e., negative phase during inhalation)
PD:	Pressure Demand	(i.e., always positive pressure)
R:	Recirculating	(closed circuit)

2/ (a) For purposes of this Specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency (above 99.9% removal efficiency by U.S. Bureau of Mines type dioctyl phthalate (DOP) test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

3/ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5/, below, concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR, Part 20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Bureau of Mines in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR, Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

C. The following maintenance and test procedures will be developed to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

1. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

2. Routine testing of standby and redundant equipment.

3. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

4. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

5. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

D. Temporary changes to procedures described in A, B, and C above, which do not change the intent of the original procedure may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented, and reviewed and approved by the Operations Committee within one month.

6.6 Plant Operating Records

A. Records and logs relative to the following items shall be retained for five years.

1. Normal plant operation including such items as power level, periods of operation at each level, fuel exposure and shutdowns.

2. Written shift supervisory and reactor logs.

3. Periodic checks, inspections, tests and calibrations of components and systems, as related to these Technical Specifications.

4. Reviews of changes made to procedures or equipment or reviews of tests and experiments.

5. Radioactive shipments

6. Abnormal occurrences

7. Principal maintenance activities, including inspection, repairs and substitution or replacement of principal items of equipment pertaining to nuclear safety.

8. Records of changes to plant procedures and records of special tests and experiments.

9. Records of wind speed and direction.

10. Records of individual plant staff members showing qualifications, training and retraining.

B. Records and logs relative to the following items shall be retained for the life of the plant.

1. Liquid and gaseous radioactive releases to the environs

2. Radiation exposures for all plant personnel

3. Off-site environmental monitoring surveys

4. Fuel accountability including new and spent fuel inventories and transfers and assembly histories

5. Plant radiation and contamination surveys

6. Changes made to the plant as it is described in the Final Safety Analysis Report, reflected in updated, corrected and as-built drawings

7. Cycling through transients for those components that have been designed to operate safely for a limited number of cycles through such transients

8. Reactor coolant system in-service inspections

9. Minutes of meetings of the Safety Audit Committee

6.7 Plant Reporting Requirements

The following information shall be submitted in addition to other applicable reports required by Title 10, Code of Federal Regulations

A. Routine Reports

Operations Reports shall be submitted in writing to the Deputy Director for Reactor Projects, Directorate of Licensing, USAEC, Washington, D C 20545.

1. Startup Report

A summary report of unit startup and power escalation testing shall be submitted following receipt of operating licenses, following amendments to the licenses involving a planned increase in power level, following the installation of fuel that has a different design and/or has been manufactured by a different fuel supplier, or following modifications to an extent that the nuclear, thermal, or hydraulic performance of the unit may be significantly altered. The report shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation should be described. Startup reports shall be submitted with 60 days following resumption of commercial power operation.

2. Semiannual Operating Reports

Semiannual operating reports shall be submitted within 60 days after January 1 and July 1 of each year and covering a six-month period. These reports should include the following:

a. Operations Summary

A summary of operating experience occurring during the reporting period that relates to the safe operation of the plant, including a summary of:

(1) Changes in plant design,

(2) performance characteristics (e.g., equipment and fuel performance),

(3) changes in procedures which were necessitated by (1) and (2) or which otherwise were required to improve the safety of plant operations,

(4) results of surveillance tests and inspections required by these Technical Specifications,

(5) the results of any periodic containment leak rate tests performed during the reporting period,

(6) a brief summary of those changes, tests, and experiments requiring authorization from the Commission pursuant to 10 CFR 50.59 (a), and

(7) any changes in the plant operating organization which involve positions which are designated as key supervisory personnel on Figure 6.1.2.

b. Power Generation

A summary of power generated during the reporting period including:

(1) Gross thermal power generated (in MWH),

(2) gross electrical power generated (in MWH),

(3) net electrical power generated (in MWH),

(4) number of hours the reactor was critical,

(5) number of hours the generator was on-line, and

(6) histogram of thermal power versus time.

c. Shutdowns

Descriptive material covering all outages occurring during the reporting period. For each outage, information should be provided on:

(1) The cause of the outage,

(2) the method of shutting down the reactor; e.g., trip, automatic rundown, or manually controlled deliberate shutdown,

(3) duration of the outage,

(4) plant status during the outage (e.g., cold shutdown or hot standby), and

(5) corrective action taken to prevent repetition, if appropriate.

d. Maintenance

A discussion of safety-related maintenance (excluding preventative maintenance) performed during the reporting period on systems and components that are designed to prevent or mitigate the consequences of postulated accidents or to prevent the release of significant amounts of radioactive material. Included in this category are systems and components which are part of the reactor coolant pressure boundary defined in 10 CFR 50.2(v), any part of the engineered safety features, or associated service and control systems that are required for the normal operation of engineered safety features, part of any reactor protection or shutdown systems, or part of any radioactive waste treatment handling and disposal system or other system which may contain significant amounts of radioactive material. For any malfunction for which corrective maintenance was required, information should be provided on:

(1) The system or component involved,

(2) the cause of the malfunction,

(3) the results and effect on safe operation,

(4) corrective action taken to prevent repetition, and

(5) precautions taken to provide for reactor safety during repair.

e. Changes, Tests, and Experiments

A summary of all changes in the plant design and procedures that relate to the safe operation of the plant shall be included in the Operations Summary section of these semiannual reports. Changes, tests, and experiments performed during the reporting period that require authorization from the Commission pursuant to 10 CFR 50.59(a) are covered in paragraph 6.7.A.2.a(6) of these Technical Specifications; however, those changes, tests, and experiments that do not require Commission authorization pursuant to 10 CFR 50.59(a) shall be addressed. The report shall include a brief description and the summary of the safety evaluation for those changes, tests, and experiments, carried out without prior Commission approval, pursuant to the requirements of 10 CFR 50.59(b) of the Commission's regulations, that "The licensee shall furnish to the Commission, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each."

f. Radioactive Effluent Releases

Effluent data should be summarized on a monthly basis following a standard format as issued by the USAEC.

(1) Gaseous Effluents

(a) Total gross radioactivity (in curies) including noble and activation gases released

(b) Maximum gross radioactivity release rate during any one-hour period

(c) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed

(d) Percent of Technical Specification limit

(2) Iodine Releases

(a) Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed

(b) Percent of Technical Specification limit for I-131 released

(3) Particulate Releases

(a) Gross radioactivity (B, γ) released (in curies) excluding background radioactivity

(b) Gross alpha radioactivity released (in curies) excluding background radioactivity

(c) Total gross radioactivity released (in curies) of nuclides with half-lives greater than eight days

(d) Percent of Technical Specification limit for particulate radioactivity with half-lives greater than eight days

(4) Liquid Effluents

(a) Gross radioactivity (B, γ) released (in curies) excluding tritium and average concentration released to the unrestricted area

(b) Total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area

(c) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area

(d) Total volume (in liters) of liquid waste released

(e) Total volume (in liters) of dilution water used prior to release from the restricted area

(f) The maximum concentration of gross radioactivity (β, γ) released to the unrestricted area (averaged over the period of release)

(g) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed

(h) Percent of Technical Specification limit and 10 CFR Part 20 concentration limits for unrestricted areas

(5) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet)

(b) The total estimated radioactivity (in curies) involved

(c) The dates and disposition (if shipped offsite)

g. Environmental Monitoring

(1) For each medium sampled, e.g., air, river bottom, surface water, soil, fish including:

(a) Number of sampling locations

(b) Total number of samples

(c) Number of locations at which levels are found to be significantly above local backgrounds

(d) Highest, lowest, and the annual average concentrations or levels of radiation for sampling point with the highest average and description of the location of that point with respect to the site

(2) If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, 10 CFR Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variations of offsite environmental concentrations with the time are observed, correlation of these results with effluent release shall be provided.

h. Occupational Personnel Radiation Exposure

A tabulation of personnel exposures shall be reported for the year (or first six months) in the following groups: less than 100 mRem, 100 - 500 mRem, 500 - 1250 mRem, 1250 - 2500 mRem, 2500 - 5000 mRem, above 5000 mRem. An explanation for all personnel exposures greater than 500 mRem in six months or the year shall be provided.

B. Non-routine Reports

1. Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone, followed by a telegram, to the Director of the AEC Region III Regulatory Operations Office, followed by a written report within 10 days to the Deputy Director for Reactor Projects, Directorate of Licensing (copy to the Director of Region III Regulatory Operations Office) in the event of an abnormal occurrence as defined in Section 1.0. The written report on an abnormal occurrence, and to the extent possible, the preliminary telephone and telegraph notification shall:

a. Describe, analyze and evaluate safety implications,

b. Outline the measures taken to assure that the cause of the condition is determined, and

c. Indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems

In addition, the written report should relate any failures or degraded performance of system and components for the incident to similar equipment failures that may have previously occurred at the plant. The evaluation of the safety implications of the incident should consider the cumulative experience obtained from the record of previous failures and malfunctions of the affected systems and components or of similar equipment.

2. Report of Unusual Events

A written report should be forwarded within 30 days to the Deputy Director for Reactor Projects, Directorate of Licensing and to the Director of the Region III Regulatory Operations Office, in the event of:

a. Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Final Safety Analysis Report or in the bases for the Technical Specifications.

b. Discovery of any substantial variance from performance specifications contained in the Technical Specifications or in the Final Safety Analysis Report.

c. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

C. Special reports shall be submitted in writing to the Deputy Director for Reactor Projects, Directorate of Licensing, USAEC, Washington, D C 20545.

1. Reports on the following areas shall be submitted as noted:

Area	Specification Reference	Submittal Date
a. Primary Containment Leak Rate Tests	4.7A	90 days after completion of each test requiring a summary technical report
b. Secondary Containment Leak Rate Tests (1)	4.7C	90 days after completion of the first refueling outage
c. In-service Inspection Evaluation, & Development	4.6F & 4.6F Bases	Five years (2)
d. Analysis of Surveillance Specimens	4.6B Bases	Within one year after removal from the vessel
e. Main Steam Line Isolation Valve Leakage	4.7A Bases	18 months (2)
f. Summary Status of Fuel	2.1 Bases	90 days after each refueling outage starting with second refueling outage
g. Failed Fuel Detection	3.2 Bases	5 years (2)
h. Primary Coolant Leakage to Drywell	4.6D Bases	18 months (2)
i. Instrument Line Flow Check Valve Evaluation	4.7D Bases	90 days after completion of first refueling outage

NOTES: (1) This summary technical report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of these data which demonstrate compliance with the specified leak rate limits. Surveillance testing at refueling outages after the first operating cycle should be reported in the semiannual operating reports.

(2) The summary technical report shall be submitted within the period of time listed based on the initial commercial service date as the starting point.

2. When it is determined that maintenance to restore components or systems covered by these Technical Specifications to an operable condition will last longer than the allowable out of service time, it shall be the subject of a special report. This report shall be submitted to the AEC for receipt prior to the end of the allowable out of service time and shall describe:

a. The nature of the problem and specific steps to be taken to remedy the situation,

b. an estimate of the time required to return the component (or system) to an operable condition,

c. the amount of component (or system) redundancy remaining or the availability of the other system(s) to perform the same function as the inoperable component (or system),

d. surveillance requirements on the operable component (or system), and

e. basis for the continuation of operation during the extended maintenance period.

Discovery of any significant changes of the information supplied in a, b, c, d, or e shall be reported within seven days.