FROM:
Northern States Power Company
Minneapolis, Minnesota 55401
D.D. Bohn

DATE OF DOCUMENT: Aug. 20, 1971

DATE RECEIVED: Aug. 31, 1971

NO.: 5679

LTR.: X

MEMO:

REPORT:

OTHER:

TO:
Dr. Peter A. Morris

ORIG.: 1

CC:

OTHER:

ACTION NECESSARY ☐	CONCURRENCE ☐	DATE ANSWERED:	
NO ACTION NECESSARY ☐	COMMENT ☐	BY:	

CLASSIF: U

POST OFFICE REG. NO:

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DESCRIPTION: (Must Be Unclassified)
Ltr trans the following:

ENCLOSURES: (1) Request for Authorization of a change (Change No. 3) in Tech Specs of App A for Lic. No. DPR-22 notarized 8-19-71....
(2) Exhibit A-describes the proposed change along w/reasons for change.....
(3) Exhibit B-copy of Tech Specs marked up to indicate the proposed changes....
(3 Orig & 19 Conf'd cys of Request &

REMARKS: 19 cys ea encl(Exhibit's A & B) rec'd)

REFERRED TO	DATE	RECEIVED BY	DATE
Knuth w/6 cys for ACTION	9-1-71		
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U.S. ATOMIC ENERGY COMMISSION

MAIL CONTROL FORM FORM AEC-326S (6-60)

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NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

August 20, 1971

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AUG 31 1971
U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

Dr Peter A Morris
Director - Division of Reactor Licensing
United States Atomic Energy Commission
Washington, DC 20545

Dear Dr Morris:

MONTICELLO NUCLEAR GENERATING PLANT E-5979
Docket No. 50-263
CHANGE REQUEST NO. 3

Attached are three signed originals and nineteen conformed copies of a request for changes in Technical Specifications of Appendix A of the Provisional Operating License DPR-22 for the Monticello Nuclear Generating Plant. This change request has been reviewed and approved by the Monticello Operations Committee and the Safety Audit Committee.

Yours very truly,

E C Ward, Director
Engineering Vice Presidential Staff

By D D Bohn
D D Bohn, P. E.
Supervising Environmental Engineer

DDB/lb

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 No. 3)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for change. Exhibit B is a copy of the Technical Specifications marked up to indicate the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By AV Dienhart
A V Dienhart
Vice President - Engineering

Subscribed and sworn to before me

this 19th day of August, 1971

Robert E. Hessian
Robert E Hessian
Notary Public, Hennepin County, Minnesota
My Commission Expires May 15, 1976

Notarial Seal

3879

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	c. Isotopic analyses including determination of tritium of representative batches of liquid effluent shall be performed and recorded at least once per quarter. Each batch of effluent released shall be counted for gross alpha and beta activity and the results recorded. At least once per month a gamma scan of representative batches of effluent shall be performed and recorded to determine the gamma energy peaks of these batches. If energy peaks other than those determined by the previous isotopic analyses are found, a new set of isotopic analyses shall be performed and recorded.
	d. Grab samples will be taken from the discharge canal monthly and analyzed for tritium and significant isotopes.
	e. ~~The liquid effluent radiation monitor shall be calibrated quarterly.~~ Delete

Exhibit B - Continued

Table 4.8.1

SAMPLE COLLECTION AND ANALYSIS
MONTICELLO NUCLEAR PLANT - ENVIRONMENTAL MONITORING PROGRAM

Type of Sample	Type of Analysis	Collection Site	Collection Frequency	Samples Per Year	Analysis Per Year
River Water	GB, GS H^3 (M), Sr^{90} (Q)	Upstream 600 ft. from in-take canal. Downstream 600 ft. from discharge canal. St. Paul raw water intake.	Weekly	156	328
Lake Water	GB, GS H-3, Sr-90	5 local lakes 1 control lake	Monthly	72	288
Well Water	GB, GS H-3, Sr-90	6 sites within 5 miles of plant site including the Monticello Well.	Quarterly	24	96
Precipitation	GS, GB, H-3 I-131, Sr-90	Meteorological Station Plant site. State Health Dept. Bldg.-Mpls.	Monthly	24	120
Lake and River Bottom Sediment	GB, GS Sr-90, Cs-137	5 local lakes, 1 control lake. Upstream of plant. Downstream of plant.	Semi-annually	16	64
Plankton, Algae or Insects	GB, GS Sr-90, Cs-137	5 local lakes, 1 control lake. Upstream of plant. Downstream of plant.	Quarterly (When Available)	32	128

Exhibit B - Continued

Table 4.8.1
Continued

Type of Sample	Type of Analysis	Collection Site	Collection Frequency	Samples Per Year	Analysis Per Year
Aquatic Vegetation	GB, GS Sr-90, Cs-137	5 local lakes, 1 control lake. Upstream of plant. Downstream of plant.	Quarterly (When Available)	32	128
Clams	GB, GS	Upstream of plant. Downstream of plant.	Quarterly (When available)	8	16
Fish	GB, GS	Upstream of plant. Downstream of plant.	Quarterly (When Available)	8	16
Milk	GS, I-131 Sr-90, Cs-137	Two farms/region, four regions.	Monthly	96	384
Topsoil	GB, GS Sr-90, Cs-137	From 3 fields downwind of plant site, also 3 fields irrigated with river water downstream of plant.	Semi-annually	12	48
Vegetation	GB, GS I-131	From 3 fields downwind of the plant site.	Semi-annually	6	18
Agricultural Crops	GB, GS, I-131 Sr-90, Cs-137	From 3 fields irrigated by river water downstream from the plant.	Annually (at harvest)	3	15

3.0 LIMITING CONDITION FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

2. Both diesel generators are operable and capable of feeding their designated 4160 volt buses.

3. A second source of off-site power (reserve transformer 1AR) is fully operational and energized to carry power to the plant 4160V ac buses.

4. (a) 4160V Buses #15 and #16 are energized.
   (b) 480V Load Centers #103 and #104 are energized.

5. All station 24/48, 125, and 250 volt batteries are charged and in service, and associated battery chargers are operable.

B. When the mode switch is in Run, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3 and 3.9.B.4. or the reactor shall be placed in a cold shutdown condition.

1. Transmission Lines

From and after the date that incoming power is available from only one line, reactor operation is permissible only during the succeeding seven days unless an additional line is sooner placed in

Exhibit B - Continued

3.0 LIMITING CONDITIONS FOR OPERATION

c. For the diesel generators to be considered operable, there shall be a minimum of 26,250 gallons of diesel fuel (7 days supply for 1 diesel generator at full load) in the diesel oil storage tank.

4. Station Battery Systems

~~From and after the date that~~ If one of the two 125 V battery systems or the 250 V battery system is made or found to be inoperable for any reason, ~~reactor operation is permissable only during the succeeding seven days~~ 24 hours ~~unless such battery systems are sooner made operable~~. an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours unless such battery systems are sooner made operable.

3.9/4.9-4

4.0 SURVEILLANCE REQUIREMENTS

c. During each refueling outage, the conditions under which the diesel generators are required will be simulated and tests conducted to demonstrate that they will start and be ready to accept the emergency load within ten seconds.

d. During the monthly generator test, the diesel fuel oil transfer pump and diesel oil service pump shall be operated.

e. Once a month the quantity of diesel fuel available shall be logged.

f. Once a month a sample of diesel fuel shall be taken and checked for quality.

4. Station Battery Systems

a. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured.

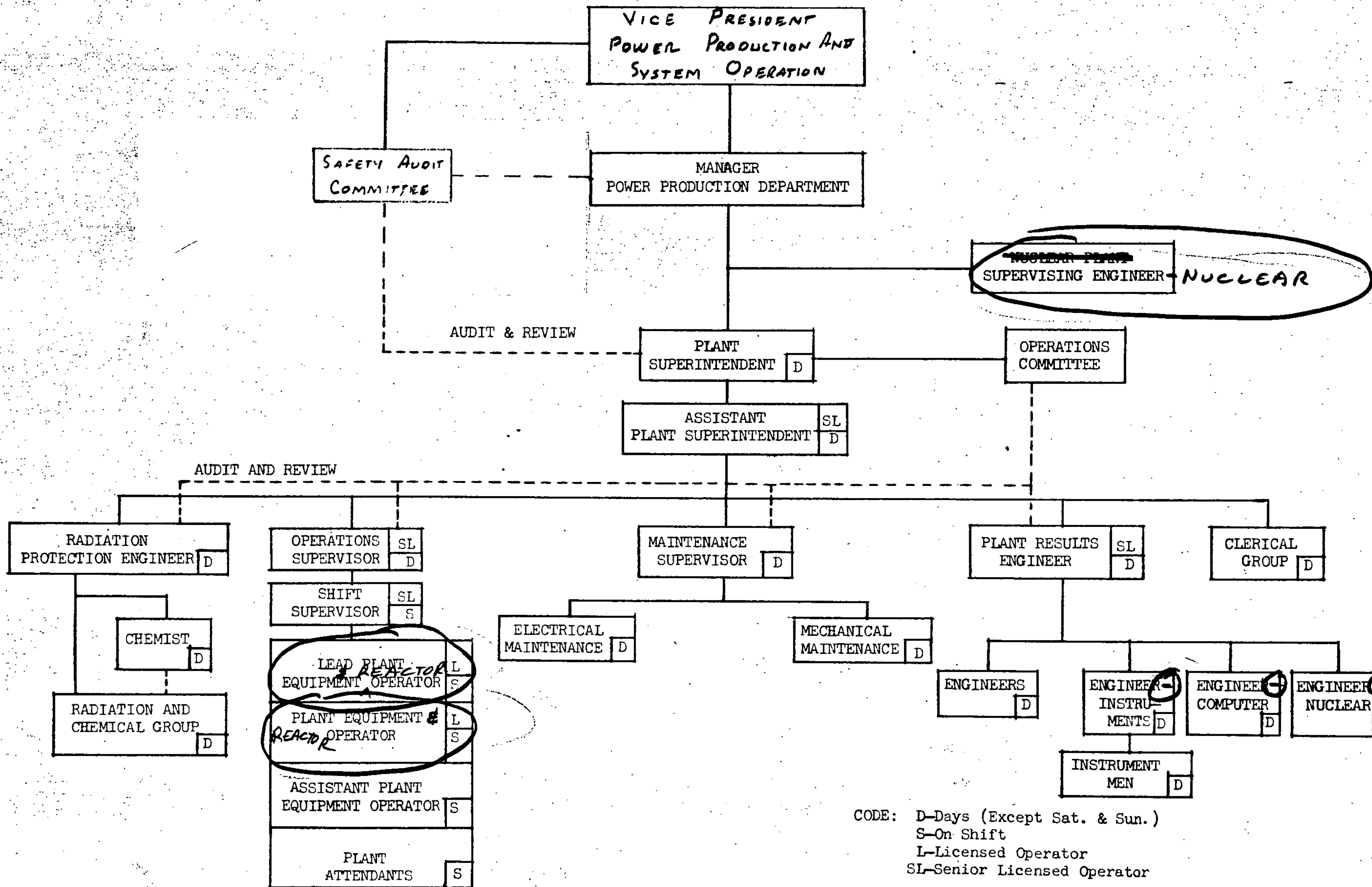

FIGURE 6.1.2 NORMAL FUNCTIONAL ORGANIZATION CHART

NSP-1

f. Records

(1) Minutes shall be recorded for all meetings of the Safety Audit Committee. The minutes shall be distributed to the Vice President-Power Production and System Operation, each member of the Safety Audit Committee, the Manager of Power Production, and others designated by the Chairman or Vice Chairman.

(2) Reports of all audits including the recommendations of the Safety Audit Committee shall be made in writing to the Vice President-Power Production and System Operation, the Manager of Power Production, the Plant Superintendent, and others designated by the Chairman or Vice Chairman.

(3) The findings of all reviews of all license and technical specification violations and recommendations to prevent recurrence shall be reported in writing to the Vice President-Power Production and System Operation, the Manager of Power Production, with copies to the Plant Superintendent.

g. Procedures — Written administrative procedures for Safety Audit Committee operations shall be prepared and maintained. These procedures shall cover the following:

(1) Content and method of submission of presentations to the Safety Audit Committee.
(2) Use of subcommittees.
(3) Review and approval, by members, of Safety Audit Committee actions.
(4) Dissemination of minutes.
(5) Detailed listing of items to be reviewed by the Safety Audit Committee.
(6) Scheduling of meetings.

2. Operations Committee

a. Membership — The Operations Committee will have at least the following members:

(1) Plant Superintendent
(2) Assistant Plant Superintendent
(3) Radiation Protection Engineer

6.3 Actions to be taken in the event of an Abnormal Occurrence in Plant Operations

A. Any abnormal occurrence shall be promptly reported to the Vice President-Power Production and System Operation or his delegated alternate and shall be promptly reviewed in accordance with the requirements specified in Table 6.1.1. A separate report shall be prepared for each abnormal occurrence. The report shall include an evaluation of the cause of the occurrence and recommendations for appropriate action to prevent or reduce the probability of a ~~repitition~~ of the occurrence.
repetition

B. Copies of all such reports shall be submitted to the Vice President-Power Production and System Operation and the Chairman of the Safety Audit Committee for review and approval of any recommendation.

C. The Plant Superintendent or Assistant Plant Superintendent shall notify the AEC within 24 hours as specified in Specification 6.6 of the circumstances of any abnormal occurrence. A written report shall follow within 10 days.

6.4 Action to be taken in the event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shutdown immediately and reactor operation shall not be resumed until authorized by the AEC and NSP management and approved by the Safety Audit Committee. An immediate report shall be made to the Vice President-Power Production and System Operation or his designated alternate, and the Chairman of the Safety Audit Committee. Notification of such occurrences will be made to the AEC by the Plant Superintendent or the Assistant Plant Superintendent within 24 hours as specified in Specification 6.6. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations to prevent a recurrence, shall be prepared by the Operations Committee. This report shall be submitted to the Vice President-Power Production and System Operation and the Safety Audit Committee.

6.5 Plant Operating Records

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years unless a longer period is required by applicable regulations.

1. Records of plant operation, including all instrumentation charts with recorded information necessary for the evaluation of reactor core performance.

Area	Specification Reference	Submitted Date
h. Primary Coolant Leakage to Drywell	4.6D Bases	18 months (3)
i. Instrument Line Flow Check Valve Evaluation	4.7D Bases	Upon completion of first refueling outage
j. Vibration Tests (4)	3.6F Bases	1 year (3)

4. A comprehensive report presenting the results of the initial preoperational, startup, power ascension, and full power test programs shall be submitted within one year of the commercial service date.

NOTES:

(1) Each integrated leak rate test of the primary containment shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report as described in the AEC Guide on Containment Testing dated ~~January 10~~ December 15, 1966, shall include data, analysis, and interpretations of the results which demonstrate compliance in meeting the specified leak rate limits.

(2) Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This 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.

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

(4) The vibration tests of the reactor internals shall be the subject of a summary technical report. This report should include:

a. The number and type and location of the vibration measuring devices used and a description of

Table 4.2.1 - Continued

NOTES:

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

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

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

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

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

** Add to Note (2) Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed as necessary, in the APRM/IRM overlap region.

Exhibit B - Continued

Bases Continued:

3.2 The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of HPCI and 300% of RCIC design flows, and valve closure times are such that core uncovery is prevented and fission product release is within 10 CFR 100 guidelines.

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

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCHFR does not decrease to 1.0. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents a signigicant reduction in MCHFR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCHFR is maintained greater than 1.0.

THE RBM provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is referenced to flow. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked when MCHFR is ~1.3, thus allowing adequate margin. Ref. Section 7.4.5.3 and 14.5.3 FSAR. Below ~70% power the worst case withdrawal of a single control rod results in a MCHFR >1.0 without rod block action, thus below this level it is not required. Requiring at least half of the normal LPRM inputs from each level to be operable assures that the RBM response will be adequate to prevent rod withdrawal errors.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCHFR approaches 1.0. Ref. Section 7.4.4.3 FSAR.

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 3/125 of full scale.

3.0 LIMITING CONDITIONS FOR OPERATION

Explode one of two primer assemblies manufactured in the same batch to verify proper function. Then install as a replacement, the second primer assembly in the explosion valve of the system tested for operation

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be

4.0 SURVEILLANCE REQUIREMENTS

systems and pump demineralized water into the reactor vessel. This test checks explosion of the charge associated with the tested system, proper operation of the valves and pump capacity. Both systems shall be tested and inspected, including each explosion valve in the course of two operating cycles.

b. ~~Explode two of four charges manufactured in same batch to verify proper function. Then install as replacement charges the untested charges in the explosion valves.~~

c. Test that the setting of the system pressure relief valves is between 1350 and 1450 psig.

B. Surveillance with Inoperable Components

When a component becomes inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter.

Exhibit B - Continued

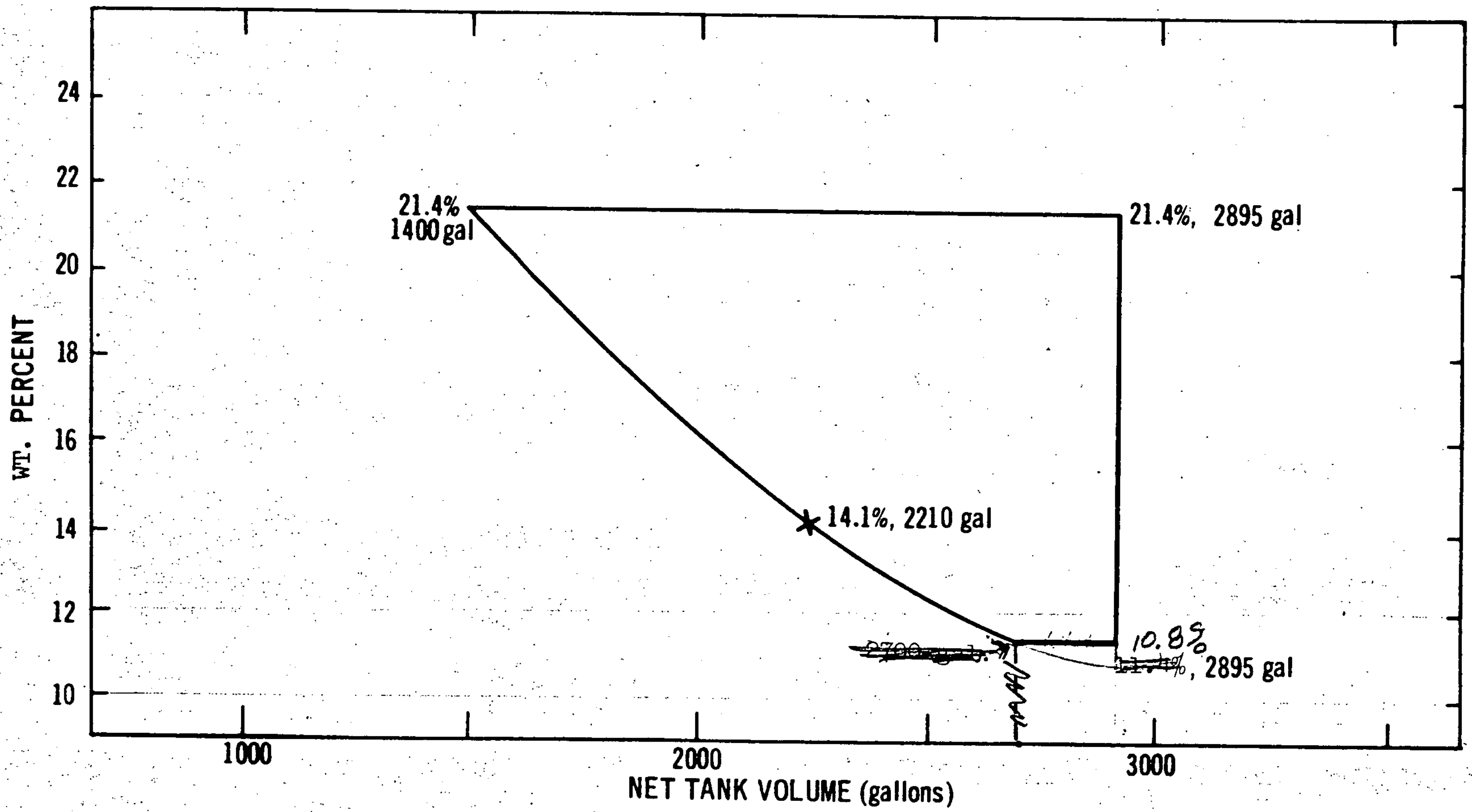

FIGURE 3.4.1. Sodium Pentaborate Solution Volume — Concentration Requirements

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. To be considered operable, the HPCI system shall meet the following conditions:	
a. The HPCI shall be capable of delivering 3,000 gpm into the reactor vessel for a reactor pressure range of ~~1150~~ 1120 psig to 150 psig.	
b. The condensate storage tanks shall contain at least 75,000 gallons of condensate water.	
c. The controls for automatic transfer of the HPCI pump suction from the condensate storage tank to the suppression chamber shall be operable.	
4. If the requirements of 3.5.D.1-2 cannot be met, either 3.5.H shall be complied with or procedures for an orderly reactor shut-down shall be initiated immediately and the reactor pressure shall be reduced to 150 psig within 24 hours thereafter.	
E. Automatic Pressure Relief System	E. Surveillance of the Automatic Pressure Relief System shall be performed as follows:

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

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

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

The performance of the reactor coolant leakage detection system,including an evaluation of the speed and sensitivity of detection, will be evaluated during the first 18 months of plant operation, and the conclusions of this evaluation will be reported to the AEC. Modifications, if required, will be performed during the first refueling outage after AEC review. In addition, other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety and Relief Valves

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. A tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the set pressure, the reactor coolant pressure safety limit of 1375 psig is not exceeded. Safety/relief valves are used to minimize activation of the safety valves. The operator will set the pressure settings at or below the settings listed. However, the actual setpoints can vary as listed in the basis of Specification 2.4.

Repeated Above

~~The operator will set the pressure settings at or below the settings listed. However, the actual setpoints can vary as listed in the basis of Specification 2.4.~~

Delete

The required safety valve steam flow capacity is determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a turbine trip initiated with the reactor at 1670 MWt. The analysis assumes no steam bypass system flow, no turbine valve trip scram, but a reactor scram from indirect means (high flux). The relief and safety valve capacity is assured to total 50% (35% relief and 15% safety) of the full power steam generator rate. This capacity corresponds to assuming that three of the four relief/safety valves (35.4%) and two of the four safety valves (18.5%) operated. For additonal margin three safety and three safety/relief valves are required to be operable.

3.0 LIMITING CONDITIONS FOR OPERATION

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

B. Standby Gas Treatment System

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

4.0 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

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

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

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

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

b. At least once during each scheduled secondary containment leak rate test, whenever a filter is changed, whenever work is performed that could affect the filter system efficiency, and at intervals not to exceed six months between refueling outages, it shall be demonstrated that:

(1) The removal efficiency of the particulate filters is not less than 99% for particulate matter larger

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	than 0.3 micron based on a dioctyl-phthalate (DOP) test. (2) The removal efficiency of the charcoal filters is not less than 99% for freon based on a freon test. c. At least once each five years removable charcoal cartridges shall be removed and adsorption shall be demonstrated. d. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
2. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit including its emergency power source shall be operable.	2. When one circuit of the standby gas treatment system becomes inoperable, the operable circuit including its emergency power source shall be demonstrated to be operable immediately. The operable circuit of the Standby Gas Treatment System shall be demonstrated to be operable daily thereafter.
3. If this condition cannot be met, procedures shall be initiated immediately to establish the conditions listed in 3.7.C.1. (a) through (d), and compliance shall be completed within 24 hours thereafter.	

Exhibit B - Continued

3.0 LIMITING CONDITIONS FOR OPERATION

C. Secondary Containment

1. Secondary containment integrity, shall be maintained during all modes of plant operation except when all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212° and the reactor coolant system is vented.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

4.0 SURVEILLANCE REQUIREMENTS

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. ~~A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain at least a 1/4 inch of water vacuum under calm wind (< 5mph) conditions with a filter train flow rate of 4,000 scfm.~~ Delete

   b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a filter train flow rate of ≤ 4,000 scfm, shall be demonstrated at each refueling outage prior to refueling.

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
APPENDIX A OF PROVISIONAL OPERATING
LICENSE NO. DPR-22

1. Page i, Item 3.2 and 4.2, C
   Change to:
   C. Control Rod Block Actuation
   Reason for Change:
   The word "Block" was an omission in previous submittals

2. Page ii, Item 3.4 and 4.5
   Change to:
   3.5 and 4.5

3. Fuel Cladding Integrity Safety Limit

   Page 7, Item 2.1 B, Change to read:
   When the reactor pressure is less than 600 psig or core flow is less than 5% of design, the reactor thermal power transferred to the coolant shall not exceed 300 MW.

   Page 10, Figure 2.1.1
   Extend the two curves linearly from the present termination at 25% to 5% at 18% thermal power (300 MW).

   Page 15, Bases: 2.1
   Change the first two sentences of the last paragraph to read:

   The range in pressure and flow used for Specification 2.1.A was 600 psig to 1250 psig and 5% to 100% flow respectively. Specification 2.1.B provides a requirement on power level when operating below 600 psig or 5% flow.

   Reason for change:
   The basis given for fuel cladding integrity safety limits applies to the lower core flow limit of 5% but with less margin than the current lower limit of 25% flow. Following recirculation pump trip test at Monticello, the operating point was found to be close to the 25% flow limit line. The change will decrease the chance of a "technical" violation of the specification and will make the safety limit curve consistent with curves presently in use at other facilities.

4. Pages 20, 21, 22, and 26
   These pages refer to deviations discussed on Page 22. The deviations are discussed on Page 18, not Page 22.

5. Page 30, Item 4.c on Table 3.1.1
   Add "limiting Trip Setting" of $\leq 3/125$ of full scale. This was an omission in previous submittals.

3879

6. Page 39, Third paragraph
   Sign on 10% is incorrect, should be ≤ 10%.

7. Page 51, Item 4 on Table 3.2.1
   Change Item 4 to be as follows:

	Function	Trip Settings	Total No. of Inst. Ch. Per Trip System	Min. No. of Operable or Operating Inst. Ch. Per Trip System (1,2)	Required Conditions
4.	HPCI Steam Lines				
a.	HPCI High Steam Flow	≤150,000 lb/hr with ≤60 sec time delay	2(4)	2	F
b.	HPCI High Steam Flow	≤300,000 lb/hr	2(4)	2	F
c.	HPCI Stm Line Area High Temp.	≤ 200°F	16(4)	16	F

Reason for change:
On HPCI initiation, initial steam flow surges cause instrumentation indications in excess of the current trip setting of 150,000 lb/hr. Testing of this system has shown that normal steam flow indications are reached within 60 seconds after initially exceeding the 150,000 lb/hr trip setting. The trip setting of 300,000 lb/hr will provide steam line break protection during the time delay period on the 150,000 lb/hr trip. Wording in the current Technical Specifications is actually in error with the trip system agreed upon with DRL and installed in the plant. This change corrects the wording to be consistent with the existing trip system.

8. Page 59, **Allowable Bypass Conditions
   Change Item d. to read:
   d. SRM Upscale block may be bypassed when associated IRM range switches are above Position 6.

Reason for change:
The SRM Upscale rod block is not necessary when the IRMs are above range 1. SRM Upscale rod blocks occur when IRM indication reaches the middle of range 7. Allowing the block to be bypassed on range 7 will eliminate the need to prematurely switch the IRMs to range 8. Also, additional margin will be provided to allow for possible future changes in IRM and SRM calibration.

9. Page 63, Note (2) of Table 4.2.1
   Change Note (2) to read:

Calibrate prior to normal shutdown and start-up and thereafter check once per shift and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.

Reason for change:
The SRM and IRM rod blocks are bypassed in the run mode. Calibration can only be carried as far as the instrument channel trip. The most accurate method of calibrating the IRM output signal is by comparison with the APRMs in the overlap region.

10. Page 67, First Paragraph
Change last sentence to read as follows:

The trip settings of 200°F, 150% of HPCI and 300% of RCIC design flows, and valve closure times are such that core uncovery is prevented and fission product release is within 10 CFR 100 guidelines.

Reason for change:
The trip setting for RCIC High Steam Flow listed on Page 51 is ≤ 45,000 lb/hr. Design steam flow is 16,500 lb/hr maximum. The 150% trip setting referenced in the original draft was meant to apply to the HPCI system only.

11. Page 89, Item 4.4.A.2.b
Change paragraph b to read as follows:

b. Explode one of two primer assemblies manufactured in the same batch to verify proper function. Then install as a replacement, the second primer assembly in the explosion valve of the system tested for operation.

Reason for change:
At the time when this specification was originally discussed with the AEC-DRL staff, it was thought that the two explosive charges associated with each valve were independent devices which could be exploded individually. Actually, the replacement charges come in the form of a single primer assembly. The primer assembly contains two charges, either of which firing alone will result in valve operation. Explosion of one primer assembly is therefore the explosion of two charges. The proposed change is consistent with specification 4.4.A.2.a, which requires testing of only one valve each operating cycle, and the present bases which states "A test of explosive charges from one manufacturing batch is made to assure that the replacement charges for the tested system are satisfactory."

12. Page 92, Figure 3.4.1
    Continue lower portion of curve to terminate at 10.8% at 2895 gallons. Remove limit of 11.4% at $\geq$2700 gallons.

    Reason for change:
    In one of the early draft versions of the Technical Specifications the bottom portion of the curve was removed to assure that the required amount of boron would be injected within 100 minutes at a minimum flow of 27 gpm. Subsequently, the requirements were changed. The present specification and bases require a minimum flow of 24 gpm and a pumping time not to exceed 125 minutes. Restoration of the lower portion of the curve is consistent with the present bases and will allow lower concentrations, thereby reducing minimum temperature requirements and reducing the possibility of crystallization.

13. Page 104, Item 3.5.D.3.a
    Change to:

    a. The HPCI shall be capable of delivering 3,000 gpm into the reactor vessel for the reactor pressure range of 1120 psig to 150 psig.

    Reason for change:
    Design specifications of 1135 psia to 165 psia were evidently miscalculated to 1150 psig instead of 1120 psig. 1120 psig is the design value and sufficient overlap with maximum reactor pressure is assured by setting the relief valves at 1080 psig shown in Spec. 4.6.E.2.a.

14. Page 134, Item 3.6.E
    Delete second paragraph.
    Reason for change:
    This wording is a repeat of last two sentences in paragraph above.

15. Page 148, Item 4.7.B
    Correct outlining by adding 1 before paragraph beginning with "Standby gas----------".

16. Page 148, Item 4.7.B.1.b
    Change first sentence to read as follows:

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

    Reason for change:
    As presently written, it requires filter tests (DOP & Freon) during each scheduled secondary containment leak rate test. It has been interpreted that this means testing must be simultaneous with leak rate testing. This testing requirement is to demonstrate removal efficiency prior to refueling operations and is irrelevant to simultaneous testing with the leak rate test. The testing requirement is satisfied with the proposed wording.

Also, we are required by Specification 4.7.C.1.b to perform additional secondary containment leak rate tests during the initial operating cycle under various wind conditions to enable valid extrapolation of leak rate test results. It is not desirable nor does it seem necessary to require the additional filter testing with the leak rate tests that are done for this reason.

17. Page 149, Item 4.7.1.c
    Correct sentence structure by adding and after removed.

18. Page 150, Item 4.7.C.1
    Delete paragraph a.
    Reason for change:
    These preoperational requirements have been satisfied. Reletter and insert present paragraph c as paragraph a.

19. Page 150, Item 4.7.C.1
    Add sign $\leq$ before 4000 scfm in new paragraph a.

    Reason for change:
    This change still requires the original integrity but allows more flexibility in operation and testing.

20. Page 172, Item 4.8.C.2.e
    Delete this paragraph.
    Reason for change:
    The requirements are a repeat from paragraph 4.8.C.1, page 171.

21. Page 174, Table 4.8.1
    Add "When Available" to Collection Frequency for Plankton, Algae or insects.

    Reason for change:
    These organisms generally do not inhabit the river in abundance necessary to do radiological analysis. For example, no samples were available during the last quarter of 1970.

22. Page 175, Table 4.8.1
    Add "When Available" to Collection Frequency for Aquatic Vegetation and for Fish.

    Reason for change:
    Vegetation growth in the river in the area of the plant exists only during part of the year. Fish sampling is limited to the middle of April to about the first week in November which eliminates the winter quarter sample.

23. Page 181, Item 3.9.B
    Add to the sentence after 3.9.B.4 the following:
    "or the reactor shall be placed in a cold shutdown condition".
    Reason for change:
    This was an omission in the previous submissions.

24. Page 183, Item 3.9.B.4
    Change to read as follows:

4. Station Battery Systems

If one of the two 125 V battery systems or the 250 V battery system is made or found to be inoperable for any reason, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours unless such battery systems are sooner made operable.

Reason for change:
The proposed change is more restrictive than existing requirements but is more consistent with intended operation.

25. Page 194, Figure 6.1.2 Normal Functional Organization Chart
Add block for Vice President - Power Production and System Operation and change advisory function of the Safety Audit Committee from Manager of Power Production and System Operation to Vice President - Power Production and System Operation. Recent title changes are also made on the revised Figure 6.1.2 in Exhibit B attached.

Reason for change:
The change will make the organization chart consistent with the wording in the 6.1.E.1.e, Page 195.

26. Page 196, Item 6.1.E.1.f.(3)
Correct sentence by adding and: - - - the Vice President - Power Production and System Operation and the Manager of Power Production, with copies to the Plant Superintendent.

27. Page 203, Item 6.3.A
Correct spelling of repetition in last sentence.

28. Page 210, Note (1)
Change date of AEC Guide to December 15, 1966. This later date is the current AEC Guide on Containment Testing.

Exhibit B

TABLE OF CONTENTS

Exhibit B - Continued

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS
	$$S = \frac{486{,}000\ P}{X}$$ Where: P = percent of rated power X = peak heat flux - ($BTU/HR/FT^2$) shall be used.
	2. IRM--Flux Scram setting shall be ≤ 15% of rated neutron flux.
B. When the reactor pressure is less than 600 psig or core flow is less than ~~25%~~ 5% of design, the reactor thermal power transferred to the coolant shall not exceed 300 MW.	B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the curve in Figure 2.3.2. When the combination of power and peak flux is above the curve in Figure 2.3.2, a rod block trip setting (RB) as given by: $$RB = \frac{437{,}400P}{X}$$ where: P = percent of rated power X = peak heat flux ($BTU/HR/FT^2$) shall be used.
C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.3.A for longer than 0.95 seconds as indicated by the process computer.	C. Reactor Low Water Level Scram setting shall be ≥ 10'6" above the top of the active fuel.

Exhibit B - Continued

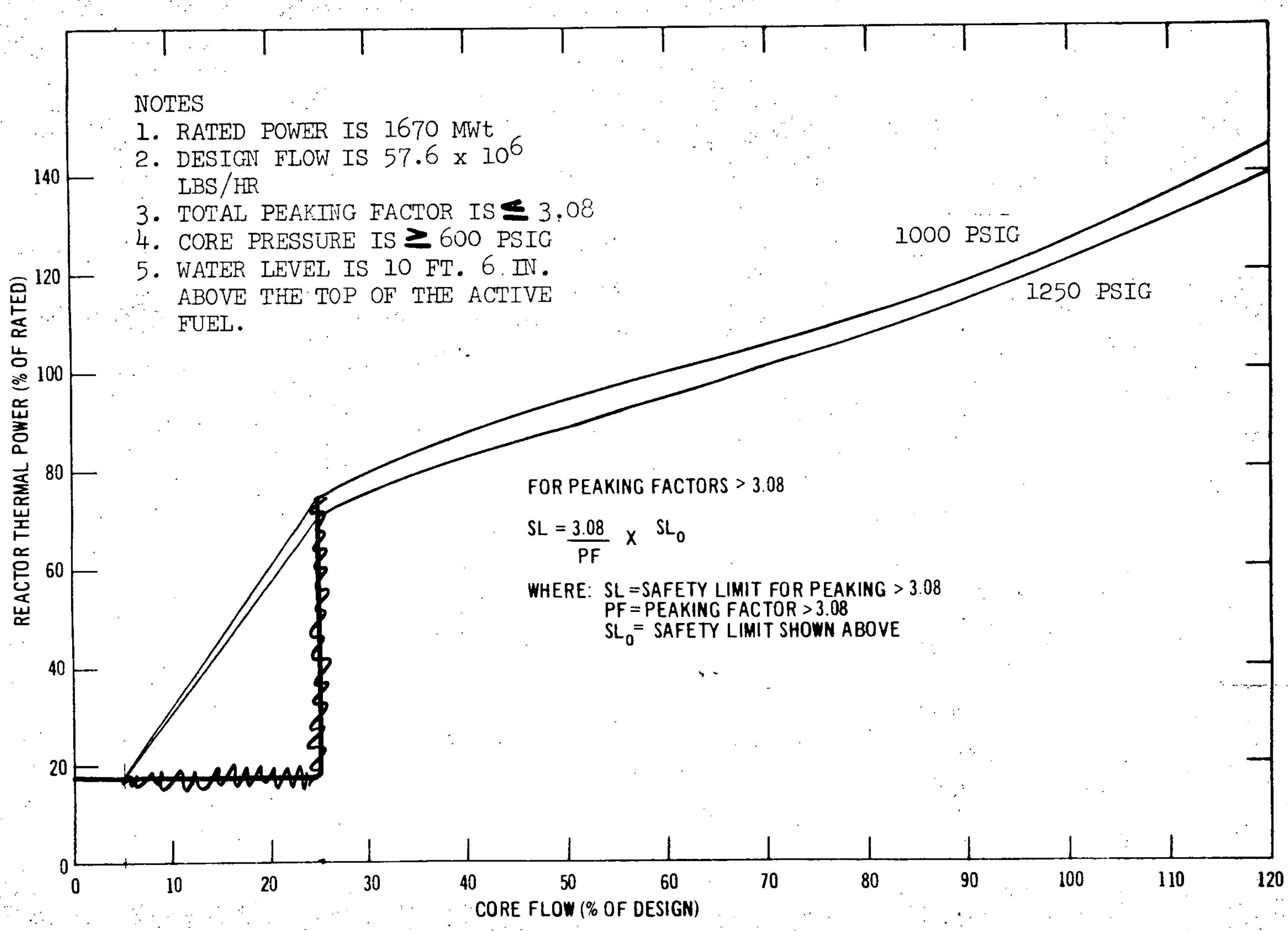

FIGURE 2.1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

Bases Continued:

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows, which is 376°F for rated thermal power. For any lower feedwater temperature, sub-cooling is increased and the curves are conservative.

The water level assumed in the calculation of the safety limit was that level corresponding to the bottom of the steam separator skirt (7" on the level instrument is equivalent to 10'6" above the top of the active fuel at rated power). As long as the water level is above this point, the safety limit curves are applicable; i.e., the amount of steam carry under would not be increased, and, therefore, the core inlet enthalpy and sub-cooling would not be influenced.

The values of the parameters involved in Figure 2.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the Average Power Range Monitor (APRM) in-core nuclear instrumentation is calibrated to read in terms of percent power.

The range in pressure and flow used for Specification 2.1.A was 600 psig to 1250 psig and ~~25%~~ 5% to 100% respectively. Specification 2.1.B provides a requirement on power level when operating below 600 psig or ~~25%~~ 5% flow. In general, Specification 2.1.B will only be applicable during startup or shutdown of the plant. A review of all the applicable low pressure and low flow data (2, 3) has shown the lowest data point for transition boiling to have a heat flux of 144,000 BTU/HR/Ft^2. To assure applicability to Monticello fuel geometry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., critical heat flux was assumed to occur for these conditions at 72,000 BTU/HR/Ft^2. Assuming a peaking factor of 3.08, this is equivalent to a core average power of approximately 300 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions, there is increased margin.

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(2) E. Janssen - "Multirod Burnout at Low Pressure" - ASME Paper 2-HT-26, August 1962.
(3) K. M. Becker - "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters" - AE-74 (Stockholm, Sweden), May, 1962.

Bases Continued:

2.3 For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 18% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

The operator will set the APRM neutron flux trip setting no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page ~~20~~ 18.

B. APRM Control Rod Block Trips - Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at a given recirculation flow rate, and thus protects against exceeding a MCHFR of 1.0. This rod block set point, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable set point provides substantial margin from fuel damage, assuming steady state operation at the set point, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship, therefore,

Bases Continued:

2.3 the worst case MCHFR during steady state operation is at 110% of rated power. Peaking factors as specified in Section 3.2 of the FSAR were considered. The total peaking factor was 3.08. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors greater than 3.08 exist. This assures a rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by changing the intercept point of the flow bias curve (keeping the slope constant); thus, the entire curve will be shifted downward.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on Page 18.

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

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual set point can be as much as 6 inches lower due to the deviations discussed on Page 18.

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

Bases Continued:

2.3 The operator will set the low low water level ECCS initiation trip setting $\geq$ 6'6" $\leq$ 6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page ~~22~~. 18.

E. Turbine Control Valve Fast Closure Scram - The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Reference Sections 14.5.1.1 and 14.5.1.2 FSAR.

F. Turbine Stop Valve Scram - The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting at 10% of valve closure, only a slight increase in surface heat flux occurs as shown in FSAR Figure 14.5.3 and thus adequate margin exists. The primary system relief valves open to limit the pressure rise, then reclose as pressure decreases. For this condition the peak surface heat flux is less than 105% of its rated power value and MCHFR remains above 1.9. Reference Section 14.5.1.2.2 FSAR.

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

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

The operator will set this pressure trip at greater than or equal to 850 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page ~~22~~. 18

Bases:

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

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety valves are sized according to the code for a condition of turbine stop valve closure while operating at 1670 MWt, followed by (1) no turbine trip valve scram, (2) failure of the turbine bypass valves to open, but (3) scram from an indirect (high flux) means. With the safety valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1293 psig. See Section 4.4.3 FSAR. Evaluations presented in the FSAR indicate that a total of five valves (2 safety valves and 3 dual purpose safety/relief valves) set at the specified pressures maintain the peak pressure during the transient within the code allowable and safety limit pressure.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 22. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1080 psig or lower. However, the actual set point can be as much as 11 psi above the 1080 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page ~~22~~ 18. #1

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

TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition
		Refuel(3)	Startup	Run			
1. Mode Switch in Shutdown		x	x	x	1	1	A
2. Manual Scram		x	x	x	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤120/125 of full scale	x	x	x(c)	4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative c. Downscale	See Specifications 2.3A.1 ≤ 3/125 of full scale			x	3	2	A or B
5. High Reactor Pressure	≤1075 psig	x	x(f)	x(f)	2	2	A
6. High Drywell Pressure	≤ 2 psig	x(4)	x(e,f)	x(e,f)	2	2	A
7. Reactor Low Water Level	≥7 in.(6)	x	x(f)	x(f)	2	2	A
8. Scram Discharge Volume High Level	≤32 gal.(8)	x(a)	x(f)	x(f)	2	2	A
9. Turbine Condenser Low Vacuum	≥23 in. Hg	x(b)	x(b,f)	x(f)	2	2	A or C

Bases Continued:

3.1 condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat iput to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass. Ref. Section 14.5.2.2 FSAR. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

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

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

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.7.1 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the

Table 3.2.1 - Continued

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions
b. High Drywell Pressure (5)	≤2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. Low Reactor Water Level	≥10'6" above the top of the active fuel	2	2	E
4. HPCI Steam Lines				
a. HPCI High Steam Flow	≤150,000 lb/hr with ≤ 60 sec time delay	2(4)	2	F
c. ~~b.~~ HPCI Steam Line Area High Temp.	≤200° F	16(4)	16	F
b. HPCI High Steam Flow	≤ 300,000 lb/hr	2(4)	2	F
5. RCIC Steam Lines				
a. RCIC High Steam Flow	≤45,000 lb/hr	2(4)	2	G
b. RCIC Steam Line Area High temp.	≤ 200° F	16(4)	16	G

Table 3.2.3 - Continued

Notes:

(6) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied actions shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(b) Place the plant under the specified required conditions using normal operating procedures.

(7) There must be a total of at least 4 operable or operating APRM channels.

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

A. Reactor in Shutdown mode.

B. No rod withdrawals permitted while in Refuel or Startup mode.

C. Reactor in Run mode.

D. No rod withdrawals permitted while in the Run mode.

E. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode.

**Allowable Bypass Conditions

a. SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel count rate is 100 cps or when all IRM range switches are above Position 2.

b. IRM Downscale rod block may be bypassed when the IRM range switch is in the lowest range position.

c. RBM Downscale rod block may be bypassed below 30% rated power.

d. SRM Upscale block may be bypassed when associated IRM range switches are above Position ~~7~~ 6.

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

RECEIVED APR 2 1971

Dr Peter A Morris
Director - Division of Reactor Licensing
United States Atomic Energy Commission
Washington, DC 20545

APR 1 1971

Dear Dr Morris:

MONTICELLO NUCLEAR GENERATING PLANT E-5979
Docket No. 50-263
CHANGE REQUEST NO. 2

Attached are a signed original and twenty conformed copies of Request for Change No. 2, covering proposed modifications to the gaseous radwaste system for the Monticello Nuclear Generating Plant.

The required technical information and related safety analysis is included in a report entitled, Gaseous Radwaste System Modification, twenty five copies of which are enclosed.

Yours very truly,

E C Ward

E C Ward, Director
Engineering Vice Presidential Staff

ECW/lb

Attachment

1670

RECEIVED
APR 2 1971
U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

UNITED STATES ATOMIC ENERGY COMMISSION

NORTHERN STATES POWER COMPANY

Monticello Nuclear Generating Plant Docket No. 50-263

REQUEST FOR AUTHORIZATION OF
MODIFICATIONS TO GASEOUS RADWASTE SYSTEM

PROVISIONAL OPERATING LICENSE NO. DPR-22

(Change Request No. 2)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the gaseous radwaste system. This request is being submitted pursuant to Section 50.59(d) of the AEC Rules and Regulations.

The required technical information and related safety analysis is included in a report entitled Gaseous Radwaste System Modification which is made a part of this request. Also included in the report are the related proposed changes to the Technical Specifications.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

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

Subscribed and sworn to before me
this APR 1 1971 day of

/s/ Robert E Hessian
Robert E Hessian
Notary Public, Hennepin County, Minnesota
My Commission Expires May 15, 1976

Notarial Seal

1670