SEP 17 1975

No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer
Director of Nuclear Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gentlemen:

The Commission has issued the enclosed Amendment No. 11 to Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment includes Change No. 19 to the Technical Specifications and is in response to your requests dated August 16, 1974, and July 1, 1975.

The amendment revises the license and appended Technical Specifications to: (1) incorporate operating limits and surveillance for the Monticello reactor vessel based on Appendix G of 10 CFR Part 50, (2) substitute a more generalized approach to the licensing of the byproduct, source and special nuclear materials and incorporate those leak testing and related surveillance and reporting requirements for the sealed radioactive sources, (3) revise specifications associated with the Augmented Off-Gas System to incorporate planned modifications to equipment and procedures, and (4) revise the radioactive iodine (131) release limits based on Regulatory Guide 1.42 and the dispersion factors calculated by the NRC staff.

Since items (3) and (4) above are not effective until the Augmented Off-Gas System is fully operational, currently estimated to be within thirty days after the Fall 1975 startup, it is requested that the Commission be notified promptly, in writing, when the system is fully operational.

Our current procedure for the licensing of byproduct, source and special nuclear materials included in reactor licenses is not to specify quantity limits. Therefore, we have issued this amendment consistent with that procedure.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 11
   w/Change No. 19
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

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Notified Mr L.C. Mayer of NSP By phone on 9-17-75 that this change has been signed

Subject to corrections & additions on pp. 4+7 of the SE.

OFFICE ▶	RL:ORB#2	RL:ORB#2	OELD	RL:ORB#2	
SURNAME ▶	RMDiggs:ah	BCBuckley	S H Lewis	DLZiemann	
DATE ▶	9/16/75	9/17/75	9/16/75	9/17/75	

-318 (Rev. 9-53) AECM 0240 GPO 643-16-81465-1 445-678

SEP 1 (1975

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Mr. D. S. Douglas, Auditor
Wright County Board of Commissioners
Buffalo, Minnesota 55313

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Secretary & Executive Officer
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Minneapolis, Minnesota 55440

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
230 South Dearborn Street
Chicago, Illinois 60604

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 11
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by the Northern States Power Company (the licensee) dated August 16, 1974, and July 1, 1975, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The receipt, possession and use of the byproduct, source and special nuclear material as authorized by this license, as amended, will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including Sections 30.33, 40.32, 70.23 and 70.31.

2. Accordingly, Provisional Operating License No. DPR-22, as amended, is hereby further amended by replacing in their entirety paragraphs 2.B., 2.C., 2.D. and 3.B. thereof with the following:

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DATE ➤						

"2.B. Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974, May 30, 1975, and July 1, 1975;

2.C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

2.D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

2.E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

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DATE ▶						

3.B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 20."

3. This license amendment is effective as of the date of its issuance, except that those Technical Specifications changes associated with the Augmented Off-Gas (AOG) System and radioiodine release limits (identified as items 3, 5 and 4 of the Northern States Power Company's application dated July 1, 1975), will not be effective until the AOG System is fully operational.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Attachment:
Change No. 19 to the
Technical Specifications

Date of Issuance: SEP 17 1975

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SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 11

CHANGE NO. 19 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the existing pages of the Technical Specifications listed below with the attached revised pages bearing the same numbers, except as otherwise noted. Changed areas on these pages are shown by marginal lines:

v
vi
48
48A
115
116
116A - addition
122
122A - addition
122B - addition
122C - addition
130
131
131A - addition
169
173
173A
176A - addition
177A
179A
179B
189B - addition
189C - addition
189D - addition
189E - addition

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DATE ➤						

☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

LIST OF FIGURES

B. Emergency Core Cooling Subsystems Actuation

When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2.

C. Control Rod Block Actuation

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

D. Air Ejector Off-Gas System

1. Except as specified in 3.2.D.2 and 3.2.D.3, both steam jet air ejector off-gas radiation monitors shall be operable during reactor power operation. The trip settings for the air ejector monitors, except as specified in 3.2.D.4, shall be set to close the recombiner train inlet valve(s) within 30 minutes at a radiation level not to exceed the equivalent of the maximum permitted stack release rate after a decay time of 120 minutes.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. From and after the date that one of the two steam jet air ejector off-gas radiation monitors is made or found to be inoperable, continued reactor power operation is permissible provided the inoperable radiation monitor instrument channel is tripped. 3. Upon loss of both steam jet air ejector off-gas radiation monitors, an orderly shut-down shall be initiated and the reactor shall be in cold shutdown within 24 hours. 4. If operation is necessary with the Off-Gas Holdup System recombiners bypassed, the trip settings for the air ejector monitors shall be reset to close the stack off-gas isolation valve within 15 minutes at a radiation level not to exceed the equivalent of the maximum permitted stack release rate after a decay time of 30 minutes.	

3.0 LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

19 A. Reactor Coolant Heatup and Cooldown

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr. when averaged over a one-hour period.

2. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the idle recirculation loop is within 50°F of the reactor coolant temperature.

4.0 SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Reactor Coolant Heatup and Cooldown

19 During heatups and cooldowns the following temperatures shall be recorded at least every 15 minutes until 3 consecutive readings at each location are within 5 °F.

a. Reactor vessel shell adjacent to shell flange.

b. Reactor vessel bottom drain.

c. Recirculation loops A and B.

d. Reactor vessel bottom head.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing, the reactor vessel shell temperatures specified in 4.6.B.1 shall be at or above the higher of the temperatures shown on the two curves of Figure 3.6.2 where the dashed curve, "RPV Beltline Region," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

2. During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 3.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Beltline Region," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 3.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Beltline Region," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

4.0 SURVEILLANCE REQUIREMENTS

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing when the vessel pressure is above 312 psig, the following temperatures shall be recorded at least every 15 minutes.

   a. Reactor vessel shell adjacent to shell flange.

   b. Reactor vessel bottom head.

2. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM E 185-66. Samples shall be withdrawn at one fourth and three fourths service life. Analysis of the first sample shall include a quantitative determination of the copper and phosphorous content.

3. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated value of neutron fluence at one fourth of the beltline shell thickness that is used to determine the NDTT shift from Figure 3.6.1.

4. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are $\geq 70^{\circ}F$.

C. Coolant Chemistry

1. The steady state radioiodine concentration in the reactor coolant shall not exceed 5 microcuries of I-131 dose equivalent per gram of water.

4. When the reactor vessel head studs are under tension and the reactor is in the Cold Shutdown Condition, the reactor vessel shell flange temperature shall be
19 permanently recorded.

C. Coolant Chemistry

1. (a) A sample of reactor coolant shall be taken at least every 96 hours and

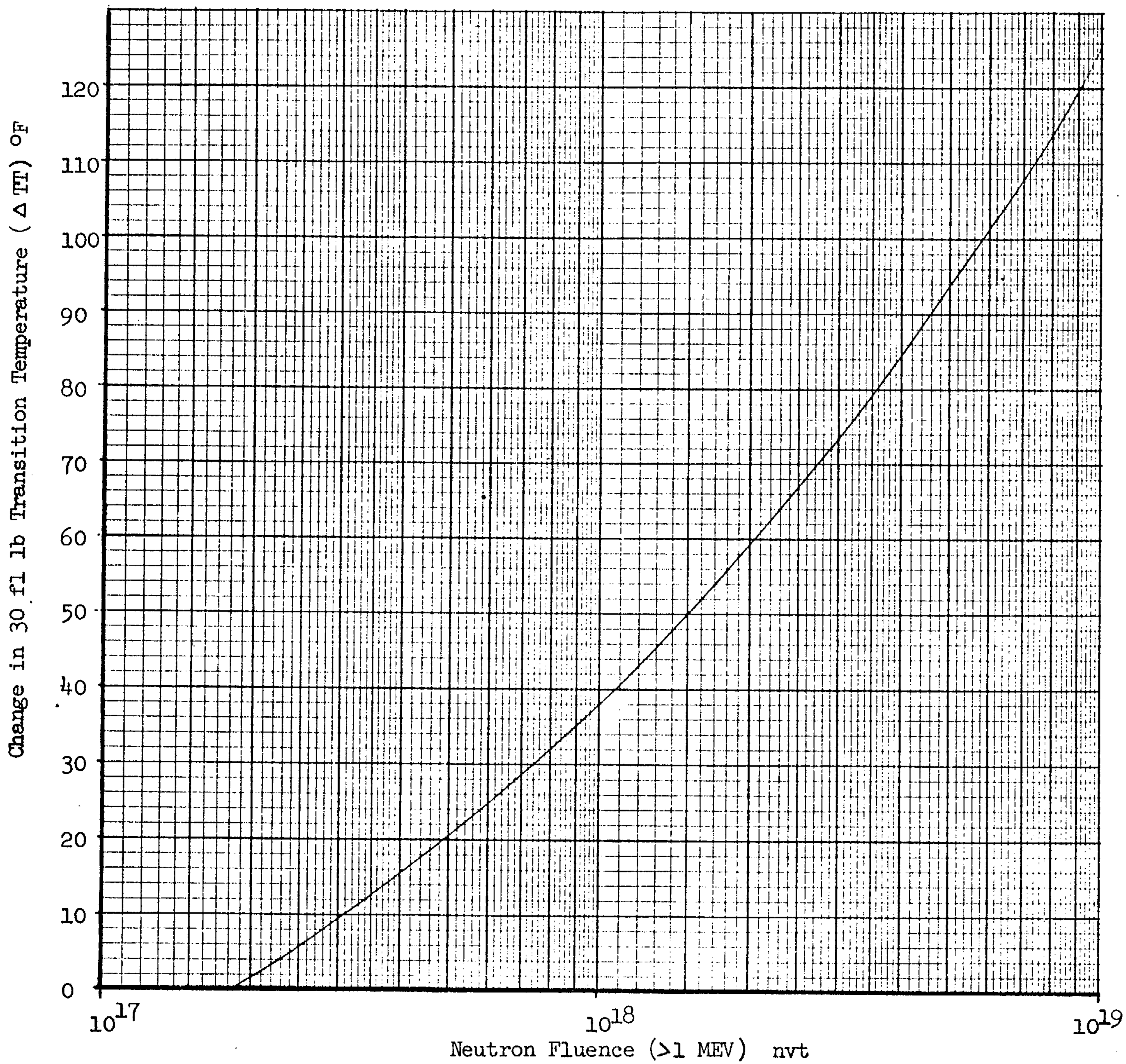

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FIGURE 3.6.1 Change in Charpy V Transition Temperature versus Neutron Exposure

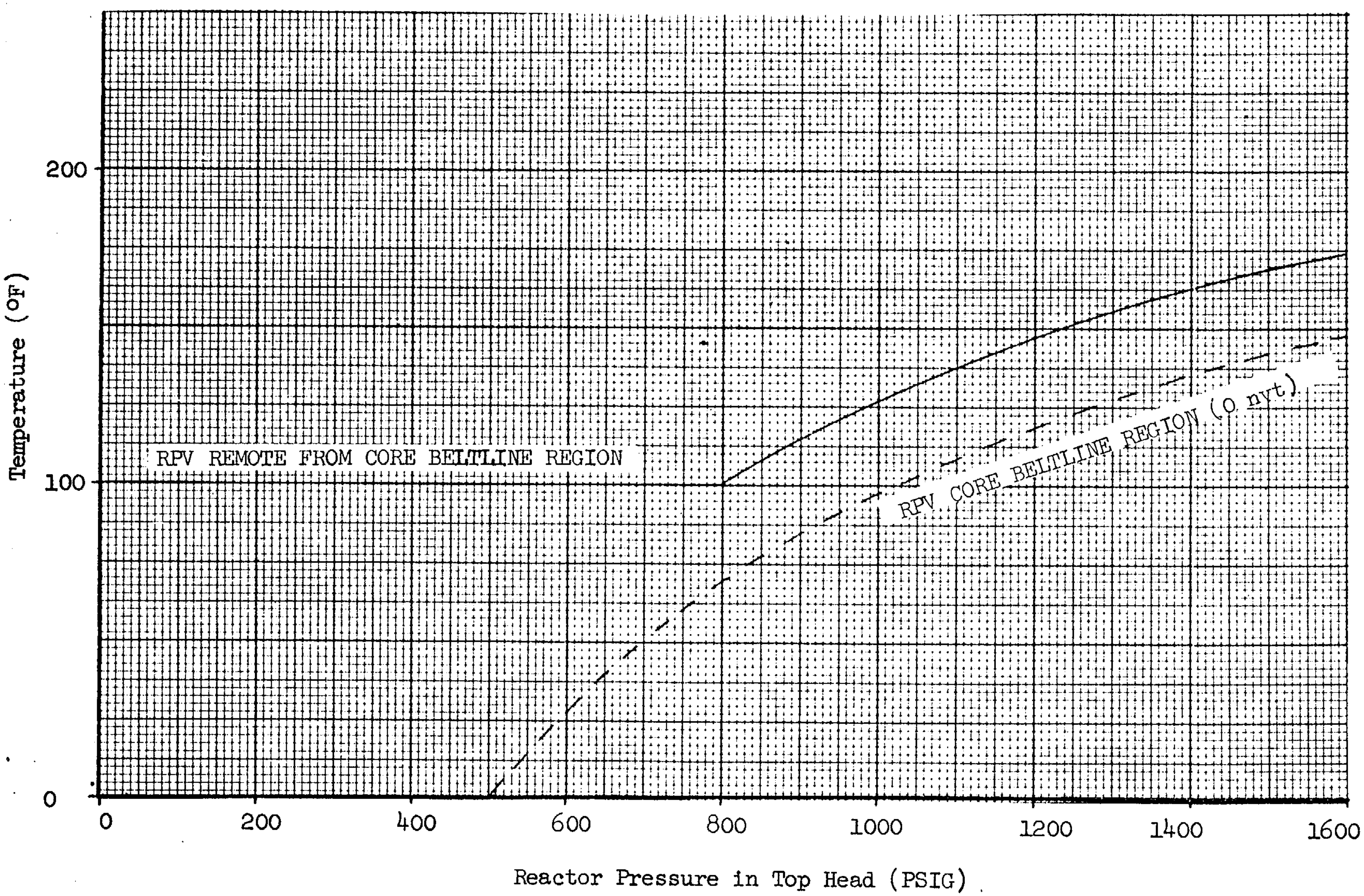


FIGURE 3.6.2 Minimum Temperature versus Pressure for Pressure Tests

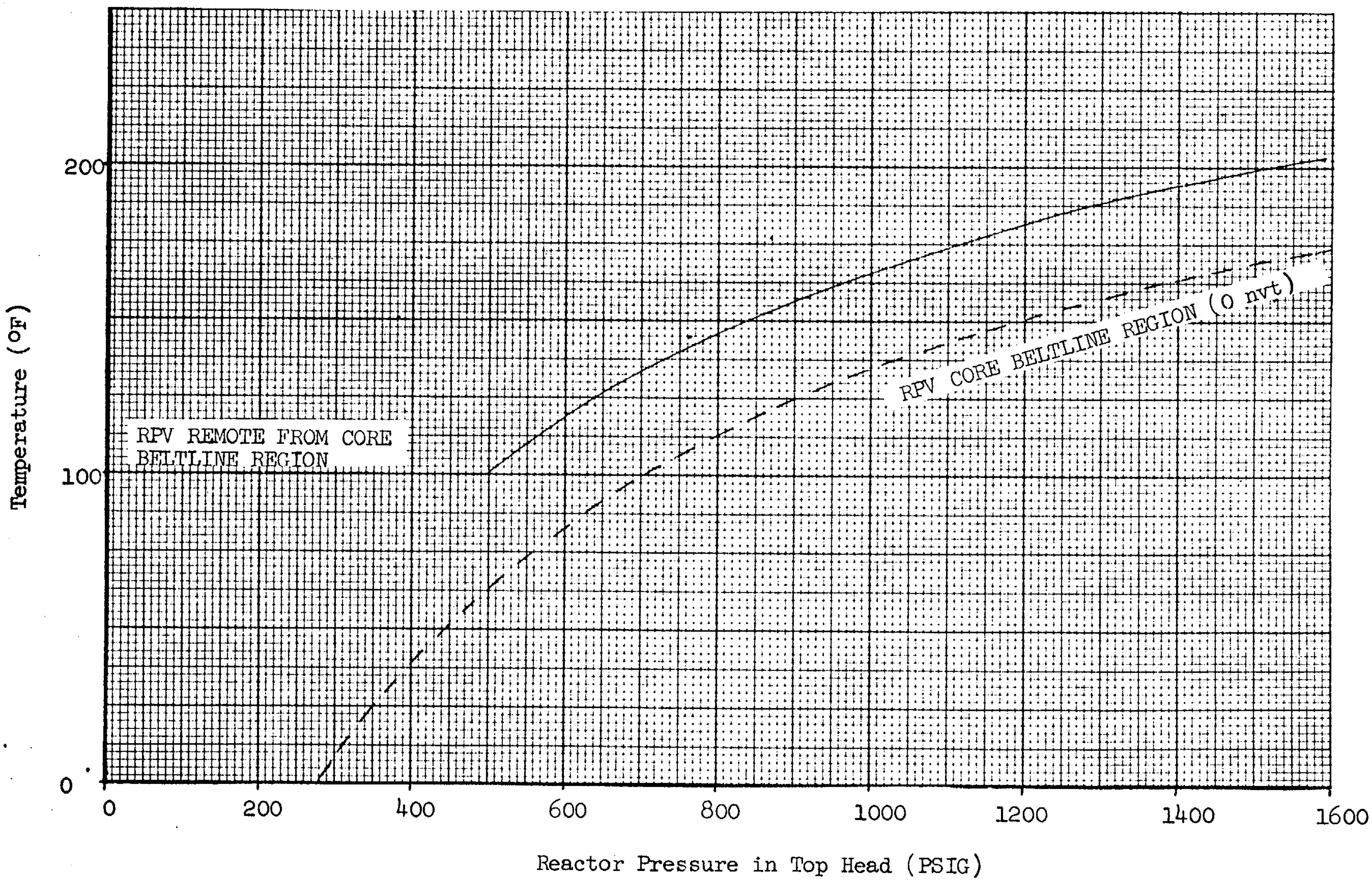


FIGURE 3.6.3 Minimum Temperature versus Pressure for Mechanical Heatup or Cooldown Following Nuclear Shutdown

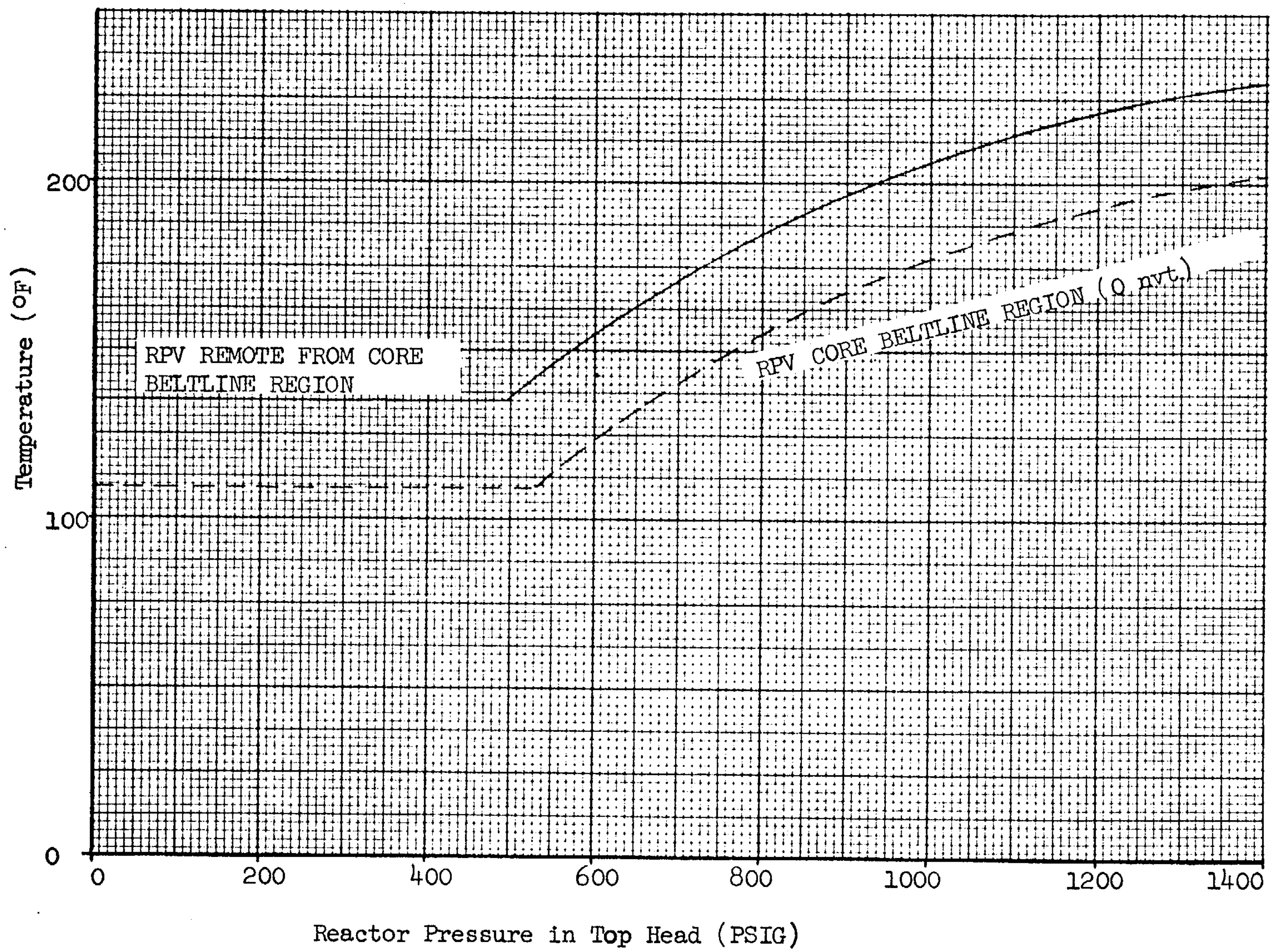


FIGURE 3.6.4 Minimum Temperature versus Pressure for Core Operation

Bases 3.6 and 4.6:

A. Reactor Coolant Heatup and Cooldown

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

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

19

B. Reactor Vessel Temperature and Pressure

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown and during inservice hydrostatic testing were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Summer 1966 Addenda). Where the dropweight NDT temperature was known, the reference temperature used was the NDT temperature. Where the dropweight NDT temperature was not known, the reference temperature used was the temperature at which 30 ftlb of energy was expected to occur on the basis of reported Charpy V notch test data. For areas of the vessel shell remote from the core beltline region, the highest NDTT permitted by the vessel purchase specification for any vessel pressure boundary material is +40°F and this value is used for the RT_{NDT} in lieu of certified test results.

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

Bases 3.6 and 4.6 - Continued:

A relationship between neutron fluence and change in Charpy V notch test 30 ftlb transition temperature has been developed for SA302B/SA533 steel based on at least 35 experimental data points as shown in Figure 3.6.1. In turn this change in transition temperature can be related to a change in the temperature ordinate shown in Figure G 2110-1 in Appendix G of Section III of the ASME Boiler and Pressure Vessel Code.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the inside diameter of the vessel wall. At present, valid experimental measurements can be made only over time periods of less than 5 years because of the limitations of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one
19 year can be used to predict the fluence for the life of the plant in thermal energy output if no great changes in core geometry are made.

The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and its relation to the neutron fluence and from Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown following nuclear shutdown), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values can be used. At the first refueling, neutron dosimeter wires which are installed adjacent to the vessel wall are removed to verify the calculated neutron fluence.

Figure 3.6.1 will be conservative for the Monticello reactor vessel. Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. These samples will receive neutron exposure more rapidly than the vessel wall and therefore will lead the vessel in integrated neutron flux exposure. An analysis and report will be submitted to the Commission on all such surveillance specimens removed from the reactor vessel in accordance with 10CFR50, Appendix H. These reports shall include the information specified in ASTM E-185-66, "Recommended Practices for Surveillance Tests on Structural Materials in Nuclear Reactors," and information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.

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

3.0 LIMITING CONDITIONS FOR OPERATION

1. The maximum release rates of gross radioactivity shall not exceed a rate Q, in curies/sec:

$$Q_1 \left(\frac{\bar{E}_\gamma}{0.18}\right) + Q_{RS} \left(\frac{\bar{E}_\gamma}{0.028} + \frac{\bar{E}_\beta}{0.019}\right) \leq 1$$

2. The release rates of gross radioactivity shall not exceed 16 percent of the limit in Specification 3.8.A.1 averaged over any calendar quarter.

3. The maximum release rate of radioiodine 131 (I-131) shall not exceed a rate Q, in microcuries/sec:

19 $$\frac{Q_1}{40} + \frac{Q_{RS}}{2.7} \leq 1$$

4. The release rate of I-131 shall not exceed 4 percent of the limit in Specification 3.8.A.3 averaged over any calendar quarter.

5. The maximum release rates of radioactive particulates with half-lives greater than 8 days shall not exceed a rate Q, in microcuries/sec:

$$\frac{Q_1}{9.5\text{x}10^9 \ \overline{MPC_a}} + \frac{Q_{RS}}{2\text{x}10^8 \ \overline{MPC_a}} \leq 1$$

where $\overline{MPC_a}$ is the composite maximum permissible concentration in air in uCi/ml determined using Appendix B, Table II, Column 1 and Notes of 10 CFR 20.

4.0 SURVEILLANCE REQUIREMENTS

1. Radioactive gases released from the off-gas stack and reactor building vent shall be continuously monitored. Station records of off-gas stack release rates of gross gaseous radioactivity shall be maintained on an hourly basis to assure that the specified rates are not being exceeded, and to yield information concerning general integrity of the fuel cladding. Records of isotopic analysis shall be maintained. The off-gas stack and reactor building vent monitoring system shall be functionally tested monthly and calibrated quarterly with an appropriate standard radiation source. Each monitor, as described, shall have a sensor check at least daily.

2. A steam jet air ejector off-gas sample shall be taken and an isotopic analysis for at least six fission product gases; Xe-138, Xe-135, Xe-133, Kr-88, Kr-85m, Kr-87 shall be made at least weekly and following each refueling or other occurrence which could alter significantly the mixture of radionuclides.

3.0 LIMITING CONDITIONS FOR OPERATION

3. Two independent samples of each tank shall be taken and analyzed for gross beta-gamma activity and the valve line-up checked prior to discharge of liquid effluents.

4. If the limits of 3.8.C cannot be met, radioactive liquid effluents shall not be released.

D. Radioactive Liquid Storage

The maximum gross radioactivity in liquid storage in the Waste Sample, Floor Drain Sample, Waste Surge, and Condensate Storage Tanks shall be less than 30 curies except for tritium and dissolved noble gases. If this condition cannot be met, the liquids in these tanks shall be recycled to tanks within the radwaste facility until the condition is met.

E. Augmented Off-Gas System

1. If the hydrogen concentration in the off-gas downstream of the recombiners reaches four percent, the recombiner off-gas flow shall be stopped automatically by closing the valves upstream of the recombiners.

2. Except as specified in Specification 3.8.E.3 below, at least one hydrogen monitor downstream of each operating
19 recombiner shall be operable during power operation.

4.0 SURVEILLANCE REQUIREMENTS

3. The performance and results of independent samples and valve checks shall be logged.

D. Radioactive Liquid Storage

1. A sample shall be taken, analyzed, and recorded within 72 hours of each addition to a liquid waste storage tank to which Specification 3.8.D. applies.

2. If the sample analysis indicates that the total radioactivity in the liquid waste storage tanks of Specification 3.8.D exceeds 30 curies, except for tritium and dissolved noble gases, the liquids in these tanks shall be recycled to reduce the radioactivity to less than 30 curies within 24 hours of this sampling.

E. Augmented Off-Gas System

1. The hydrogen monitors shall be functionally tested monthly and calibrated quarterly with an appropriate gas mixture source. Each monitor shall have a sensor check at least daily.

2. Condenser air inleakage shall be evaluated weekly and used in conjunction with the latest steam jet air ejector
19 off-gas isotopic analysis and Figure 4.8.1 to determine that the limit of Specification 3.8.E.4 will not be exceeded.

3. If the above specified downstream hydrogen monitors are not operable, offgas flow to the compressed storage subsystem shall be terminated.

19

4. The maximum gross radioactivity contained in one gas decay tank after 12 hours hold-up that can be discharged directly to the environs shall be less than 22,000 curies of Xe-133 dose equivalent. If these conditions cannot be met, the stored radioactive gas shall be recycled within 24 hours to other gas decay tanks until the condition is met.

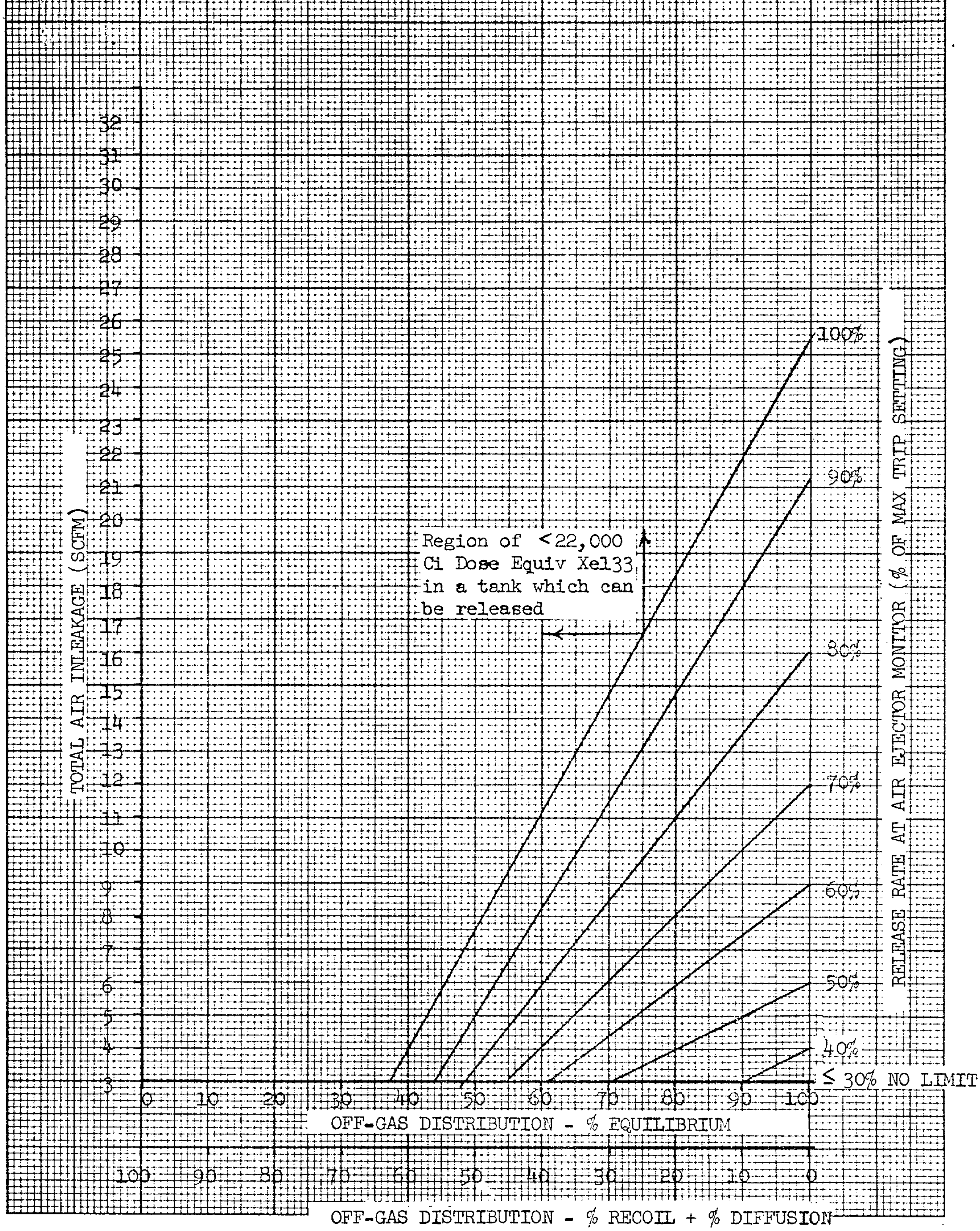
5. During normal plant operation, radioactive gaseous waste shall have a minimum holdup of 12 hours except for low radioactivity gaseous waste resulting from purge and fill operations associated with refueling and reactor startup. Holdup times for radioactive gaseous waste in the gas decay tanks shall be maximized consistent with plant operation.

19

F. Environmental Monitoring Program

The environmental monitoring program given in Table 4.8.1 shall be conducted.

Figure 4.8. Off-Gas Storage Tank Gross Activity Limits



Bases Continued:

Detailed meteorological calculations for several locations off site have been made by the AEC staff and the most critical 22.5° sector was determined to be at 600 m to the south-southeast at the site boundary. The annual average diffusion parameter value for the off-gas stack release was determined to be 1.5 x 10^{-7} sec/m^3 and for the reactor building vent release to be 7.2 x 10^{-6} sec/m^3.

The method utilized by the staff to determine annual thyroid dose of 1500 mRem to a child for I-131 releases from the off-gas stack and the reactor building vent is given in Regulatory Guide 1.42. Based on this method, the maximum I-131 concentration in milk from an existing cow would occur in the NNE sector at a distance of 3700 m which has an annual average diffusion parameter value of 2.5×10^{-8} sec/m^3 for the off-gas stack
19 and 4.3×10^{-7} sec/m^3 for the reactor building vent. Taking into account the five month grazing season, a release rate of I-131 from the off-gas stack of 40 uCi/sec or from the reactor building vent of 2.7 uCi/sec could result in an annual thyroid dose of 1500 mRem to a child drinking this milk.

In order to limit I-131 releases in the gaseous effluents to as low as practical, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and plant actions at 4 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 4, respectively, above the as low as practical objective of 1 percent of 10 CFR Part 20 limits.

The AEC staff performed an analysis similar to that used to determine the maximum release rate of I-131 for the radioactive particulates with half-lives greater than 8 days. A reduction factor of 700 on the MPC_a to allow for possible ecological chain effects similar to those associated with the cow-milk-child thyroid for radioiodine was used. The annual average diffusion parameters at 600 m in the south-southeast sector given previously were used for both the off-gas stack and reactor building vent releases. Based on these calculations, a continuous release rate of radioactive particulates with half-lives greater than 8 days in the amount of 9.5 x 10^9 $\overline{MPC_a}$ uCi/sec from the off-gas stack or 2 x 10^8 $\overline{MPC_a}$ uCi/sec from the reactor building vent would not result in annual organ doses in excess of the limits specified in 10 CFR Part 20.

In order to limit radioactive particulate releases in gaseous effluents to as low as practical, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and plant actions at 8 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 8, respectively, above the as low as practical objectives of 1 percent of 10 CFR Part 20 limits.

Bases continued:

Each batch to be released will conform to 10 CFR Part 20 release limits on an instantaneous basis, i.e., annual averaging will not be used as permitted by 10 CFR Part 20. See Section 9.2.3 of the FSAR. The radioactivity level in the discharge canal for a given release of waste will be the highest when the discharge canal flow is lowest. This occurs during "closed cycle" cooling tower operation at which time the cooling tower blowdown of approximately 36 cubic feet per second is the major flow in the discharge canal. The rate of pumping the radwaste effluent into the discharge canal is variable and can, therefore, be controlled to maintain the concentration within the specified limit. This type of operation will be employed only when the river flow is very low and will result in further dilution between discharge canal effluent and the river.

D. Radioactive Liquid Storage

The waste sample, floor drain sample, waste surge, and condensate storage tanks are not contained in a Class I structure. The maximum gross radioactivity in liquid storage in the specified tanks has been limited on the basis of an accidental spill from all stated tanks due to a seismic event great enough to damage them. Assuming a low recorded river flow of 1000 ft^3/sec, a day period over which the radioactive liquid wastes are diluted in the river, and consumption of the water by individuals at standard man consumption rate (3000 ml/day), the single intake by an individual would not exceed one-third the yearly intake allowable by 10 CFR Part 20 for unidentified radioisotopes (1×10^{-7} uCi/ml). The factor of 3 was applied to 10 CFR Part 20 limits as recommended for situations in which population groups could be exposed.

The sampling frequency has been established so that if the maximum amount of gross radioactivity is exceeded, action can be taken to reduce the radioactivity to a level below the specified limit.

E. Augmented Off-Gas System

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system.

Experience has shown that a daily check with monthly testing and quarterly calibration assures proper operation
9 of the hydrogen monitors.

The maximum gross radioactivity in one gas decay tank has been limited on the basis that accidental release of its contents to the environs by operator error after 12 hours decay should not result in exceeding the dose equivalent to the maximum quarterly release rate specified in Specification 3.8.A.2. Staff analysis of an elevated release under accident meteorology for a minimum release period of 8 hours indicated a release of 22,000 curies of Xe-133 or the dose equivalent would result in a whole body dose of 20 mRem at the nearest site boundary.

Bases Continued:

Calculations have been performed to determine the relationship between steam jet air ejector off-gas acitivity and composition and condenser air inleakage. These calculations were used to determine the curves presented in Figure 4.8.1. The results of the weekly measurement of condenser air inleakage and the average daily air ejector
19 off-gas release rate are used in conjunction with the most recent off-gas isotopic analysis to determine if the maximum permitted Xe-133 dose equivalent tank radioactivity contents may be exceeded. Daily analysis is adequate to determine that if the maximum amount of gross activity in a decay tank may be exceeded, action can be taken to reduce the radioactivity to a level below the specified limit.

F. Environmental Monitoring Program

It is recognized that a precise determination of environmental dose from a certain emission from the stack is only possible by direct measurement. Such information will be provided by the environmental monitoring program conducted at and around the site. If the stack emission ever reaches a level such that it is measureable in the environment, such measurements will provide a basis for adjusting the proposed stack limit long before the effect in the environment is of any concern for permissible dose. In this regard, it is important to realize that averaging emission rate over a period of one calendar year as permitted by 10 CFR Part 20 represents a very large safety margin between conditions at any one instant (any minute, hour, or day) and the long-term dose of interest.

3.0 LIMITING CONDITIONS FOR OPERATION

3.11 SEALED SOURCE CONTAMINATION

Applicability:

Applies to each sealed source containing more than 0.1 microcurie of plutonium or other special nuclear material (including alpha radiation) and to each sealed source containing more than the exempt quantities of byproduct materials listed in 10CFR30.71.

Objective:

To assure that leakage from sealed sources containing byproduct and special nuclear radioactive materials does not exceed allowable limits.

Specification:

A. Contamination

1. Each sealed source shall be free of removable contamination in excess of 0.005 microcuries per 100% smear test.

4.0 SURVEILLANCE REQUIREMENTS

4.11 SEALED SOURCE CONTAMINATION

Applicability:

Applies to the periodic testing of sealed sources containing more than 0.1 microcurie of plutonium or other special nuclear material (including alpha radiation) and to each sealed source containing more than the exempt quantities of byproduct materials listed in 10CFR30.71.

Objective:

To verify the leak tightness of sealed radioactive sources.

Specification:

A. Contamination

1. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

2. Each sealed source with removable contamination in excess of the limit in 3.11.A.1 shall be immediately withdrawn from use and:

   a. Either decontaminated and repaired, or

   b. Disposed of in accordance with the regulations of the Commission

a. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

b. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources exempted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested for leakage.

c. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	2. The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material per 100% smear test of the sample. B. Records 1. A complete inventory of radioactive materials in possession shall be maintained current at all times. 2. The following records shall be retained for two years: a. Test results in microcuries, for tests performed pursuant to 4.11.A. b. Record of annual physical inventory verifying accountability of sources on record.

Bases 3.11 and 4.11:

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

Small quantities of byproduct materials are exempt from licensing by 10CFR30.18 and therefore are exempt from leakage tests in these specifications. Inhalation or injestion of such small quantities of byproduct materials from a sealed source would result in less than one maximum permissible body burden for total body irradiation. Sources containing less than 0.1 microcurie of plutonium are exempt from leakage tests by 10CFR70.39(c) and therefore such quantities of special nuclear materials (including alpha emitters) are exempt from leakage tests in these specifications. The acceptance criteria of less than 0.005 microcurie on the test sample is also based on 10CFR70.39(c).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO LICENSE NO. DPR-22

(CHANGE NO. 19 TO THE TECHNICAL SPECIFICATIONS)

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

INTRODUCTION

By letters dated August 16, 1974 and July 1, 1975, Northern States Power (NSP) proposed a license amendment to Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment involves revisions to the Technical Specifications with regard to:

(1) incorporation of operating limits and surveillance for the Monticello reactor vessel based on Appendix G of 10 CFR Part 50,

(2) substitution of a more generalized approach to the licensing of the byproduct, source and special nuclear materials and incorporate those leak testing and related surveillance and reporting requirements for the sealed radioactive sources,

(3) revision of specifications associated with the Augmented Off-Gas System to incorporate planned modifications to equipment and procedures to be implemented within thirty days after the Fall 1975 startup, and

(4) revision of the radioactive iodine (131) release limits based on Regulatory Guide 1.42 and the dispersion factors calculated by the NRC staff. Such revisions would be effective when the modifications to the Augmented Off-Gas System are complete and the system determined to be fully operational.

Our evaluation of each of these subjects follows.

EVALUATION

1. Reactor Coolant System Pressure-Temperature Limitations

The current pressure-temperature limitations for operation of Monticello are based on NDT temperature plus 60°F and do not fully comply with all the requirements of Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements." The proposed pressure-temperature operating limits are based on the requirements of Appendix G, 10 CFR Part 50 and Appendix G to ASME Code Section III. In calcuations to determine these limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Summer 1966 Addenda). Where the dropweight NDT temperature was known, the reference temperature used was the NDT temperature. Where the dropweight NDT temperature was not known, the reference temperature used was the temperature at which 30 ft-lb of energy was expected to occur on the basis of reported Charpy V notch test data. For areas of the vessel shell remote from the core beltline region, the highest NDT temperature permitted by the vessel purchase specification for any vessel pressure boundary material is +40°F and this value is used for the RT_{NDT} in lieu of certified test results.

Predicted changes in NDT temperature as a function of neutron fluence are given in Figure 3.6.1. of the Technical Specifications. This curve is based on 35 data points from tests on SA 302B and SA 533B steel. It agrees with our prediction for SA 533B steel with 0.15% copper. The percent of copper in the Monticello reactor vessel plate material from the beltline region has not been determined.

Calculations indicate that the maximum neutron fluence on the vessel wall is presently about 2×10^{17} n/cm^2 and will be approximately 2.2×10^{18} n/cm^2 at end of life.

The material surveillance program for Monticello consists of three sets of specimens representing the vessel base, weld and heat affected zone (HAZ) material and conforms to ASTM E 185-66. Northern States Power Company's proposed change to withdraw samples at 1/4 and 3/4 of service life is acceptable to the staff.

We conclude that the proposed temperature-pressure limits, as specified in Figures 3.6.2, 3.6.3, and 3.6.4 of the proposed Technical Specifications filed with the application dated July 1, 1975, for operation of Monticello comply with the requirements of Appendix G, 10 CFR Part 50 and are acceptable until data from the first material surveillance capsule are obtained and reported to the NRC. We require, however, that the phosphorous and copper content of vessel plate and weld material in the vessel core region be determined at that time, and the results included in the report. This requirement has been discussed with the licensee and is acceptable. We also conclude that the proposed changes in surveillance requirements, specifications 4.5.A and B, are acceptable.

2. Byproduct, Source, and Special Material Requirements

By letter dated June 17, 1974, we requested NSP to provide the following information with regard to the Monticello Nuclear Generating Plant: (1) proposed amendments to the conditions of existing Provisional Operating License No. DPR-22 to provide more encompassing limits for the byproduct, source and special nuclear materials which NSP may receive, possess and use in connection with the operation of the facility; (2) proposed Technical Specifications for leakage testing and the related surveillance and reporting requirements for sealed radioactive material sources; (3) update their full-term license application to include the information set forth in Regulatory Guide 1.70.3 entitled, "Additional Information - Radioactive Materials Safety for Nuclear Power Plants," dated February, 1974.

The objective of the requests made in our letter of June 17, 1974 was to add flexibility to the operation of nuclear power plants by extablishing a more generalized approach to the licensing of byproduct, source, and special nuclear materials. This objective would reduce the number of licensing actions required as a result of changes in possession limits of related materials. To assure that adequate safeguards be maintained within the framework of this more generalized approach, provisions for more stringent control, accountability, and leakage testing of byproduct, source and special nuclear materials are being included in the Technical Specifications for the facility.

NSP's letter of August 16, 1974, was submitted in response to our June 17, 1974 letter and later supplemented by NSP's July 1, 1975 submittal. Since the information necessary for our review has been filed, the NRC staff has elected to act thereon now in lieu of awaiting completion of consideration of the full-term operating license application.

The proposed Technical Specification changes have been reviewed by the NRC staff with particular attention to the Radioactive Materials Safety program. We evaluated the personnel qualifications, facilities, equipment, and procedures for handling byproduct, source, and special nuclear material, as described in the August 16, 1974 application and we conclude that they are consistent with the provisions of Regulatory Guide 1.70.3. Based on our review, we also conclude that the comprehensive testing and surveillance program, as established by the proposed Technical Specification changes, provides additional assurance that leakage from radioactive material sources will not exceed allowable limits.

We further conclude that the proposed license amendment incorporating provisions relating to leak testing of sealed sources, and their inventory, storage and disposal is acceptable in that it:

a. Complies with the guidance and intent of our letter of June 17, 1974.

b. Provides reasonable assurance that byproduct, source, and special nuclear material will be stored, used, and accounted for in a manner which meets the applicable radiation protection provisions of 10 CFR Parts 20, 30, 40, and 70.

The licensee's radiation protection program, as supplemented by the proposed Technical Specifications additions, has been evaluated. We have concluded that the incorporation of flexible yet controlled licensing provisions for the receipt, possession, and use of byproduct, source, and special nuclear material into the Provisional Operating License for Monticello Nuclear Generating Plant is acceptable. This amendment to the Provisional Operating License does not authorize an increase in the amount of special nuclear material as reactor fuel.

3. <u>Air Ejector Off-Gas System</u>

The Technical Specifications currently require that the air ejector monitor trip setting be less than the equivalent of the maximum permissible stack release rate based on a 30-minute decay period. The 30-minute decay criterion is valid only when the recombiner system is in the bypass mode and is overly restrictive when the recombiner system is in operation. When only the recombiner system is in operation, the decay period ranges from 2 to 10 hours; when the compressed storage tanks are available, the decay period is approximately 50 to 250 hours. Therefore, we conclude that the 30-minute decay criterion is applicable only when the recombiner system is isolated and should be increased to 120 minutes when the recombiner system is in use and that the proposed changes to specifications 3.2.D.1 and 3.2.D.4 to reflect the above rationale are acceptable.

Item No. 5 of the July 1 application proposes (1) revisions to Specifications 3.8.E.2, 3.8.E.3, 4.8.E.2 and 4.8.E.3, (2) incorporation of a new Figure 4.8.1, "Off-Gas Storage Tank Gross Activity Limits," and (3) revisions to 3/4.8.E Bases to reflect the changes in item (1). The changes in item (1) are discussed individually below.

a. Specification 3.8.E.2

At present this specification requires that hydrogen monitors located upstream of the recombiner be operable during power operation. The licensee's proposed change would revise this requirement to monitor the hydrogen concentration downstream of the recombiner. There are three hydrogen monitors located downstream of each recombiner which would alert the operator if the hydrogen concentration exceeded 1% and would automatically isolate the recombiner system if any two of the three monitors indicate a hydrogen concentration in excess of 2%, or if any monitor indicates a hydrogen concentration in excess of 4%. The principal purpose of the hydrogen monitors is to protect the compressed gas storage tanks from a hydrogen detonation since these tanks are not designed to withstand the internal pressure that would be developed by a hydrogen detonation. All piping, valves, instrumentation and components other than the compressed storage tank system are designed to withstand a hydrogen detonation. We conclude that the proposed revision regarding monitoring of hydrogen concentration downstream of the recombiner provides appropriate protection against hydrogen detonation of the compressed storage gas system and is acceptable.

b. Specification 3.8.E.3

This existing specification requires initiation of an orderly reactor shutdown if the hydrogen monitors located downstream of the recombiner are inoperable. As discussed in (a) above, these monitors provide for protection of the compressed gas storage system and need not be operable if the compressed gas storage system is inoperable or isolated. Therefore, we have concluded that the existing specification is overly restrictive and termination of flow to the compressed gas storage system in the event all hydrogen monitors are inoperable is an acceptable precaution and the reactor need not be shut down.

c. Specification 4.8.E.2

During startup testing of the augmented off-gas system, it was determined that the compressed gas storage tank radiation monitors did not meet the design objective of measuring the gross activity of the tank contents for the following reasons:

(1) The radiation monitors are exposed to "shine" from adjacent storage tanks which defeats the intended function of monitoring the gross activity of a specific tank.

(2) The individual monitors become saturated as a result of buildup of radioactive particulates such as Rb-88 and Cs-138 and do not respond to changes in the noble gas inventory of the tank.

In addition, grab samples of the tank inventory do not provide a representative sample due to stratification within the tank. The licensee's proposed revision includes monitoring the total system air inleakage and measuring the average air ejector noble gas release rate in conjunction with Figure 4.8.1. We have reviewed and evaluated the methodology used to develop Figure 4.8.1 and find it acceptable and conclude that this revision provides reasonable assurance that the technical specification limit of 22,000 Curie dose equivalent I-133 tank inventory is not exceeded and therefore is acceptable.

d. Specification 4.8.E.3

This existing specification requires sampling and analysis of the compressed gas storage tank contents in the event the tank radiation monitor is inoperable. As discussed in (c) above, since a representative sample cannot be obtained and an alternate method of determining the tank content is available, we have concluded that deletion of this specification will not reduce the safety of operation and therefore is acceptable.

e. Figure 4.8.1 "Off-Gas Storage Tank Gross Activity"

This change consists of incorporating Figure 4.8.1 into the Technical Specifications which we found to be acceptable in (c) above.

f. Specification 3/4.8.E Bases

The Bases have been updated to reflect the above changes (a) through (e) inclusive.

4. Radioactive Iodine Limits

There have been on-going discussions between NRC and NSP with regard to the equation to be used to determine the maximum release rate of radioiodine 131 and the appropriate time when the equation would be incorporated into the Technical Specifications. We have concluded that the proposed equation conforms with Regulatory Guide 1.42 "Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors" dated March, 1974, and the dispersion factors calculated by NRC. This change would become effective when modifications to the augmented off-gas system are complete and the system has been determined to be fully operational.

We have re-evaluated the critical pathway with regard to radioiodine release and concur with the licensee that the farm located 3700 meters from the site in the NNE sector constitutes the critical pathway. We conclude that the proposed changes are acceptable.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: SEP 17 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 11 to Provisional Operating License No. DPR-22 issued to the Northern States Power Company (the licensee) for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective as of its date of issuance, except as noted in the next paragraph of this notice.

The amendment revises the license and appended Technical Specifications to: (1) incorporate operating limits and surveillance for the Monticello reactor vessel based on Appendix G of 10CFR Part 50, (2) substitute a more generalized approach to the licensing of the byproduct, source and special nuclear materials and incorporate leak testing and related surveillance and reporting requirements for the sealed radioactive sources, (3) revise specifications associated with the Augmented Off-Gas System to incorporate planned modifications to equipment and procedures to be implemented within thirty days after the Fall 1975 startup, and (4) revise the radioactive iodine (131) release limits based on Regulatory Guide 1.42 ("Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water Cooled Nuclear Power Reactors") and the dispersion factors calculated by the NRC staff. Item (4) would be effective when the modifications to the Augmented Off-Gas System are complete and the system determined to be fully operational.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the applications for amendment dated August 16, 1974 and July 1, 1975, (2) Amendment No. 11 to License No. DPR-22, with Change No. 19, and (3) the Commission's concurrently issued Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at The Environmental Conservation Library, 300 Nicollet Mall, Minneapolis, Minnesota 55414. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 17th day of September, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

OFFICE ➤						
SURNAME ➤						
DATE ➤						

The proposed Technical Specification changes have been reviewed by the NRC staff with particular attention to the Radioactive Materials Safety program. We evaluated the personnel qualifications, facilities, equipment, and procedures for handling byproduct, source, and special nuclear material, as described in the August 16, 1974 application and we conclude that they are consistent with the provisions of Regulatory Guide 1.70.3. Based on our review, we also conclude that the comprehensive testing and surveillance program, as established by the proposed Technical Specification changes, provides additional assurance that leakage from radioactive material sources will not exceed allowable limits.

We further conclude that the proposed license amendment incorporating provisions relating to leak testing of sealed sources, and their inventory, storage and disposal is acceptable in that it:

a. Complies with the guidance and intent of our letter of June 17, 1974.

b. Provides reasonable assurance that byproduct, source, and special nuclear material will be stored, used, and accounted for in a manner which meets the applicable radiation protection provisions of 10 CFR Parts 20, 30, 40, and 70.

The licensee's radiation protection program, as supplemented by the proposed Technical Specifications additions, has been evaluated. We have concluded that the incorporation of flexible yet controlled licensing provisions for the receipt, possession, and use of byproduct, source, and special nuclear material into the Provisional Operating License for Monticello Nuclear Generating Plant is acceptable. ~~These~~ This ~~change~~ amendment to the Provisional Operating License does not

3. Air Ejector Off-Gas System

The Technical Specifications currently require that the air ejector monitor trip setting be less than the equivalent of the maximum permissible stack release rate based on a 30-minute decay period. The 30-minute decay criterion is valid only when the recombiner system is in the bypass mode and is overly restrictive when the recombiner system is in operation. When only the recombiner system is in operation, the decay period ranges from 2 to 10 hours; when the compressed storage tanks are available, the decay period is approximately 50 to 250 hours. Therefore, we conclude that the 30-minute decay criterion is applicable only when the recombiner system is isolated and should be increased to 120 minutes when the recombiner system is in use and that the proposed changes to specifications 3.2.D.1 and 3.2.D.4 to reflect the above rationale are acceptable.

authorize an increase in the amount of special nuclear material as reactor fuel.

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Date: