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Docket No. 50-263

7/16/69

Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Attention: Mr. D. F. McElroy
Vice President - Engineering

Gentlemen:

On the basis of our review of your application for Unit 1 of the Monticello Nuclear Generating Plant, we find that additional information will be necessary to complete your application for a provisional operating license.

The specific information required is described in the enclosure. Most of the requested information was discussed with your representatives at meetings held on April 1-2, 1969. We recognize that some of the information requested may be available in the public record in the context of our regulatory review of similar features of other facilities. If such is the case, you may wish to incorporate the information by reference in your application.

Please contact us if you have any questions regarding this request.

Sincerely,

Called McElroy
M 7/15

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
List of Addl. Info. Required

cc: Mr. Gerald Charnoff
Shaw, Pittman, Potts, Trowbridge and Madden
910 - 17th Street, N.W.
Washington, D. C. 20006

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Alan Cabell

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-263

July 16, 1969

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414 Nicollet Mall
Minneapolis, Minnesota 55401

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Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
List of Addl. Info. Required

ADDITIONAL INFORMATION REQUIRED

MONTICELLO NUCLEAR GENERATING PLANT

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

1.0 SITE

1.1 Considering the population distribution data presented in Figures 2-2-3, 2-2-4, and 2-2-5 of the FSAR, state the low population zone radius for the Monticello facility, and give the basis for the selection.

1.2 Provide a figure which delineates the boundaries of the exclusion zone and presents, to scale, the physical features of the facility and distances to the exclusion zone boundaries. The figure should include the shortest distance from the stack and other potential release points to the exclusion zone boundary.

1.3 In regard to plant design water level, provide the following:

a. The predicted flood discharge flow and flood level at the site resulting from the maximum probable flood as defined by the U. S. Army Corps of Engineers (Policies and Procedures Pertaining to Determination of Spillway Capacities and Freeboard Allowances for Dams, Engineer Circular No. 1110-2-27, Inclosure 2, August 1, 1966, Department of the Army, Office of the Chief of Engineers).

b. The flood protection level for all Class I equipment and structures.

c. Identify any special provisions that will be provided to achieve flood protection.

1.4 Discuss the extent to which your environmental monitoring program will be capable of assessing the daily intake of radioactive material from air, water, or food by a suitable sample of an exposed population group, as related to 10 CFR 20.106(e) which states that "in addition to limiting concentrations in effluent streams, the Commission may limit quantities of radioactive materials released in air or water during a specified period of time if it appears that the daily intake of radioactive material from air, water, or food by a suitable sample of an exposed population group, averaged over a period not exceeding one year, would otherwise exceed the daily intake resulting from continuous exposure to air or water containing one-third the concentration of radioactive materials specified in Appendix "B", Table II of this part."

2.0 RADIOACTIVE WASTE CONTROL SYSTEMS

2.1 It is stated in the FSAR that the concentrations of liquid radioactive wastes released from the condenser discharge canal will be less than those permitted by the 10 CFR 20 levels on a batch-by-batch basis prior to discharge into the Mississippi River. The amount of radioactivity in the discharge will vary according to the mode of operation of the cooling towers. Based upon the various modes of cooling tower operation with the appropriate liquid radwaste release rates and river flows, calculate the radioactivity levels that would occur at the Minneapolis and St. Paul public water intakes and the intakes to the irrigational system due to the release of liquid wastes at the 10 CFR 20 levels on a batch-by-batch basis at the condenser discharge outfall. Describe the river dilution assumptions used in this analysis.

2.2 In regard to the information presented in Question 3.4 of Amendment 4 to the FDSAR concerning the storage capacity and the length of time municipal water supplies downriver from Monticello can be suspended, indicate whether this is currently applicable. If not, update this information.

2.3 At the Monticello plant there are several potential release points for airborne radioactivity; e.g., the plant stack, the reactor building vent, the cooling towers, and possibly other points. Accordingly, provide the following:

a. A list of all points within the facility where radioactivity could be released to the atmosphere.

b. The monitoring devices at each point, with alarm and isolation settings.

c. The amounts of radioactivity that could be released from each point, assuming operation at just below the isolation setting of the monitors.

d. The manner in which the releases from each of the potential release points will be factored into the routine gaseous effluent limit for the facility.

e. The basis for amounts of radioactivity, monitor set points and method of arriving at the joint effluent limit.

2.4 With respect to the various plant radioactive liquid storage tanks, provide the following:

a. The maximum radioactivity level on an isotopic basis of the maximum volume of water that can be contained in the condensate water storage tanks. Indicate the basis and assumptions used in predicting this radioactivity level.

b. The basis for the maximum inventories of liquid radioactive wastes listed in Table 9-2-1 of the FSAR.

c. A listing by isotope of the radioactivity contained within tanks in the turbine and radwaste buildings.

d. An analysis of the potential radioactivity levels in the river, at the plant site, at the irrigation system inlet, and at the Minneapolis and St. Paul public water supply intakes, that could result from the release of the contents of all of the above tanks to the river during low river water flow conditions.

2.5 The stack effluent release rate limit given in the proposed Technical Specifications is based on a specific isotopic mixture of the effluent. Recent experience with the operation of the Humboldt Bay reactor has indicated that the stack effluent isotopic mixture may change during operation of the plant. In this regard, provide the following:

a. Discuss the basis for the isotopic mixture used in arriving at the stack effluent release rate given in the proposed Technical Specifications.

b. Explain how the potential change in isotopic mixture will be factored into your analysis for establishing a stack effluent release rate.

2.6 In regard to the sampling procedures for the stack gas effluents, describe the methods that will be used to take the samples, and transport the samples to the monitors. Include in the description the following:

a. An explanation of how you will ensure representative sampling using an isokinetic sampling technique, considering that the stack gas velocity may vary with time.

b. The length of sampling line.

c. The method for preventing plateout of halogens and particulates in the sampling lines or, alternatively, the manner in which plateout will be accounted for in the analysis.

d. The frequency of removing the sampling filter and charcoal cartridge, and the method of analyzing these filters and cartridges.

e. A discussion of the adequacy of the planned sampling and monitoring methods to ensure compliance with a stack effluent release rate limit for short time periods for halogens and particulates with half-lives longer than eight days.

2.7 As stated in the FSAR, the design basis of the radwaste system and the shielding for many portions of the facility are based upon a routine gaseous release rate of 0.3 Ci/sec. However, a gaseous release limit of 0.48 Ci/sec is given in the proposed Technical Specifications. Discuss the basis for the higher operating limit, and explain why it is different from the 0.3 Ci/sec release rate given in the FSAR.

2.8 Specification 4.6 E in your proposed Technical Specifications states that ". . .grab samples will be taken from the discharge canal monthly and analyzed for tritium and significant isotopes". Explain the apparent inconsistency of this with the statement made in Section 9.2.3.1 of the FSAR that ". . .the liquid radwaste system for Monticello is designed to discharge radioactive materials within the limits of 10 CFR 20 without the annual averaging provision"; i.e., on a batch-by-batch basis.

3.0 REACTOR AND REACTOR COOLANT SYSTEM

3.1 The original application for the construction permit stated that initial operation was to be at a rated power of 1469 Mwt. This application further stated that to provide margin which would assure achieving this objective, the plant is designed to a capability of 1670 Mwt. However, your application for a provisional operating license is for 1670 Mwt. To show that these margins have not been substantially reduced, additional information is required on reactor operating characteristics and fuel limits. In this regard, provide the following:

a. The proposed fuel operating conditions for Monticello include linear power generation rates and exposures higher than those experienced previously in BWR plants. A review of available experimental data* indicates inadequate justification to support the combinations of fuel linear power generation rates and exposures requested considering both normal and anticipated transient modes of operation. Please discuss this matter in detail including plans for developing sufficient data and consideration of operating limits that may be included in the Technical Specifications for assurance of fuel integrity under normal and anticipated transient modes of plant operation. (*APED-5608 and Amendment 22 for the Oyster Creek facility).

b. Determine how close the fuel maximum centerline temperature will come to the melting point of UO_2 as a result of expected transients during the fuel lifetime. In this regard, discuss the effect of fuel exposure on the UO_2 melting point, and relate the errors involved in physics burnup calculations to errors in determining the fuel maximum centerline temperature for the exposure producing this calculated maximum temperature.

c. In Amendment 7 to the Dresden 2 and 3 application (Docket No. 50-237) responses to AEC questions 4.1 through 4.10 are addressed to various aspects of reactor performance characteristics. Discuss the extent to which this information is applicable to Monticello.

3.2 In Amendment 8 to the Dresden 2 and 3 application (Docket Nos. 50-237/249), responses to questions 7.1 and 7.2 are directed to thermal shock on reactor components induced by operation of the emergency core cooling system. Discuss the extent to which this information is applicable to Monticello. Also, provide any new information developed since submittal of the Dresden amendment.

3.3 For the Monticello steam flow restrictor nozzles, provide a drawing showing the nozzle affixed to the steam line, and an evaluation of the design including materials used, fabrication methods, applicable codes and the potential for flow-induced vibration.

3.4 With respect to the reactor vessel, provide the following information:

a. A list of stainless steel component parts in the reactor vessel and associated reactor coolant systems that have become furnace sensitized during the fabrication cycle. Include specification, grade, condition and vendor.

b. A summary of results of drop-weight tests for vessel plates.

c. State whether UT examination and mapping were performed on the control rod drive housing field welds, and provide a summary of the results.

d. Describe the extent to which UT examination and mapping of the pressure vessel was performed, and provide a summary of the results.

3.5 We recognize that during the design of the Monticello plant the proposed ASME N-45 Code for Inservice Inspection of Nuclear Reactor Coolant Systems (ASME ISI-Code) was not available for guidance. Nevertheless, an inservice inspection program for Monticello should be developed by adopting, insofar as practical, the principles and intent embodied in the ASME-ISI Code. Accordingly, provide the following information:

a. Delineate the "system boundary of the Monticello reactor coolant system"* using Figures A, B, C, and D of the ASME-ISI Code as an aid.

b. Identify the system's principal pressure-containing components and piping within the system boundary which are subject to inspection, using Table 1 of Section ISI-260 of the Code.

c. Determine the degree of accessibility to the specific location of each area (as identified under Code Section ISI-251) by detailed review of plant drawings (or by direct inspection of the facility).

d. Establish the areas and extent of examination, degree of examination sampling, and examination methods which can be used in performing the inspection for each category of components and piping specified in the Code.

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*The "system boundary of the reactor coolant system" as defined in the ASME-ISI Code comprises (a) the reactor coolant system, (b) portions of the reactor coolant associated auxiliary systems, (c) portions of emergency core cooling systems, and (d) portions of the main steam and feedwater systems.

e. Summarize in tabular form the inservice inspection which will be performed based on the application of direct or remote examination methods. The table should include the components and areas to be inspected, the sampling selection, the extent of examination, the inspection method, and the inspection frequency.

f. Describe your plans for inservice inspection of the main steam lines and emergency core cooling system lines external to primary containment.

3.6 With regard to the reactor internal structures, provide the following information:

a. A discussion of the vibration analyses that have been made which take into account both the normal and emergency modes of operation.

b. A listing of the reactor internal vibration tests which will be conducted for the Monticello plant. In this listing, include the number, type, and location of instruments for each test.

c. For test conditions other than hot functional, state the basis and methods for correlating the test data with the hot functional condition; e.g., relate the significance of changes in fluid density, and changes in clearances due to thermal expansion.

d. An evaluation of the limitations of the proposed instrumentation.

e. A discussion of what provisions will be made to take into account long-term effects by means of instrumenting and monitoring for vibration or for the presence of loose parts in the reactor pressure vessel as well as in other portions of the primary system.

3.7 Provide the summary technical information required by code in accordance with ASME Section III, N910.2 and a discussion of the margin between the peak allowable pressure and the peak vessel pressure for the transient postulated in Section 4.4.3 but assuming only a high pressure scram. For both the pressure scram and the flux scran transients, show that the assumptions used in the calculations are conservative with respect to the proposed Technical Specifications for safety system settings (Sections 2.2 and 2.3), control rod insertion times (Section 3.2.B.3), and the number of rods valved out of service (Section 3.2.B.4).

3.8 It is noted in Section 4, pages 4-2.5 and 4-3.9 of the FSAR that about 400 samples of pressure vessel material (base metal, weld zone metal, heat-affected zone metal and standard specimens) are located within the reactor vessel for periodic monitoring of material properties with irradiation exposure. In this regard, provide additional information on these samples, including location within the vessel, the method of attachment, and the means of identifying them. Describe the program for the removal and testing of these samples. Also evaluate your surveillance program relative to the ASTM E 185-66 "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors".

3.9 Describe the length of time the poison control blades are expected to remain in service, and the bases for establishing a replacement schedule for these control blades. Also, discuss the manner in which property changes of materials and the buildup of activation products, such as tritium, are considered in arriving at a useful lifetime of the poison control blades.

3.10 Since surface-carburized jet pump castings are more susceptible to cracking, discuss the safety implications involved in using this type of jet pump casting.

4.0 ENGINEERED SAFETY FEATURES

4.1 We are evaluating the adequacy of the Monticello containment and engineered safety features, using the assumptions of TID-14844 with regard to the fission product source term as a basis. To permit us to complete our review in this area, we require the following:

a. An assessment, substantiated by analyses, of the capability of the containment and engineered safety features to function in a post-accident environment, using the following assumptions:

(1) Fission products assumed to be in the recirculated water (for purposes of calculating doses on equipment and materials, radiolytic hydrogen evolution, and chemistry of solution) shall be 50% of the core halogen inventory and 1% of the core solid fission product inventory.

(2) Fission products assumed to be in the primary containment atmosphere (for purposes of calculating radiological doses, heat loads on filters, required fission product cleanup capabilities, range of radiation monitors, and radiation dose to equipment in the containment atmosphere) shall be 25% of the core halogen inventory, 100% of the core noble gas inventory, and 1% of the core solid fission product inventory. The highest expected filter efficiency shall be assumed for the purposes of calculating heat load on the filters and the lowest expected value for calculating radiological consequences. Conservative values for mixing shall be used in the secondary containment.

(3) The most conservative of the above assumptions shall be used for purposes of shielding calculations for the loss-of-coolant accident.

In your assessment include considerations relating to the standby gas treatment filter system, control room shielding and ventilation system, emergency core cooling system components, containment electrical penetrations, and materials within the containment whose decomposition or corrosion could affect the operation of vital systems.

b. If the design basis of the as-built systems is significantly different from that derived from the assumptions in part "a", show the differences between the design-basis capability, the as-built capability and the capability that would be required using the assumptions in part "a".

c. Where this assessment is sensitive to containment leakage rate, complementary analyses (using the assumptions in part "a") should be made for assumed leakage rates of 0.5% per day and 2% per day, in addition to the 5% per day leakage rate proposed in your application.

d. Are there any systems in the primary containment or reactor building necessary for post-LOCA operation that would not be expected to operate in a post-accident environment based on the assumptions in part "a"?

4.2 We have reviewed the loss-of-coolant accident analysis (Section 6.2.7 of the FSAR), and conclude that there is sufficient experimental evidence to justify your use of the level swell model to predict coolant level during blowdown. However, on the basis of our evaluation of the current state of technology, we do not believe that the use of a transient critical heat flux calculation and a transient core flow calculation is sufficiently conservative, particularly since these models have not been verified under blowdown conditions. We find that the level swell model is justified only if the General Electric dryout cooling model is used for core heatup calculations.

Considering the above comments, provide a description of, and the results of a re-analysis of the loss-of-coolant accident without using the transient critical heat flux and transient core flow calculational models. Also, provide a quantitative assessment of the analytical conservatism retained in the core heatup calculation for intermediate and large breaks (0.1 to 5.5 ft^2). Include considerations of such things as the previous critical heat flux calculation, transition and film boiling following DNB, steam cooling, residual water level, and steam availability for the metal-water reaction.

4.3 In regard to the depressurization performance of the HPCI system for intermediate break size loss-of-coolant accidents, determine the minimum required "mixing efficiency" for the operation of the HPCI system in conjunction with the LPCI system. Relate this to the peak clad temperature, considering the comments made in question 4.2 on the unacceptability of using transient critical heat flux and transient core flow calculational models.

4.4 It appears from the drawings in the FSAR that a break in the RCIC steam feed line could result in loss of HPCI capability due to the proximity of systems; i.e., the "high area temperature" signal would isolate both systems. Describe the likelihood and consequences of this event.

4.5 A bubble rise velocity of 1 ft/sec is cited on page 6-2.35 of the FSAR of the liquid break analysis while a velocity of 2 ft/sec is cited on page 6-2.40 for the steam break analysis. The 1 ft/sec assumption is less conservative for determining the time to uncover the core, while the 2 ft/sec is less conservative for determining maximum level swell in the vicinity of the steam line nozzles. Give the reason for using the two velocities and discuss the conservatism of your selection in each case.

4.6 With respect to the core spray and HPCI systems, it is noted on pages 6-2.3 and 6-2.22 of the FSAR that ". . .design of the piping system external to the reactor vessel reflects considerations for potential damage to the piping. The pipe runs of each system are physically separated and located to take maximum advantage of protection afforded by the reactor building structure." However, accidents for which protection will be required have not been identified. In this regard, provide the following information:

a. Diagrams which show the location of all emergency core cooling system piping and other essential piping systems protected by the reactor building structure.

b. The design bases for protection; i.e., for internally generated missiles, tornado generated missiles, or other accidents.

c. An evaluation describing the degree of protection afforded by the piping arrangement.

5.0 INSTRUMENTATION

5.1 Table 1-6-2 of the FSAR lists plant systems which differ from the designs of Dresden, Units 2 and 3, and Millstone, Unit 1. Extend this comparison to include the identification and evaluation of the differences in the Monticello instrumentation designs and/or design criteria from those of Dresden 2 and 3 for each of the systems: (a) reactor protection system, (b) reactor containment and reactor vessel isolation control system, (c) emergency core cooling and automatic relief systems, (d) neutron monitoring system including the RBM subsystem, (e) main steam radiation monitoring system, (f) refueling interlocks, (g) reactor manual control system, (h) reactor vessel instrumentation, (i) recirculation control system, (j) feedwater control, (k) process radiation monitoring, (l) area radiation monitoring, (m) process computer system, (n) rod worth minimizer, (o) primary leak detection, (p) turbine-generator, (q) instrument and service air systems, and (r) standby gas treatment system.

5.2 Identify the portions of those instrumentation and control systems listed in question 5.1 which are the prime design responsibility of the nuclear-steam supplier and the architect-engineer. Describe the means you have developed to ensure the implementation of design requirements for these systems.

5.3 As stated on page 3-5.18, the electrical system which actuates the directional control valves is designed such that no single failure can produce accidental movement of more than one control rod. Describe the analysis which supports this design basis.

5.4 On page 5-2.9 it states that Table 5-2-3 is a <u>typical</u> listing of principal isolation valves. Provide a listing of the isolation valves which is directly applicable to Monticello.

5.5 According to Figure 6-2-4, a core spray pump is energized after a ten-second time delay permissive is actuated. Describe and evaluate the core spray control system to show that the design is compatible with the starting and loading sequence of the diesel generator.

5.6 Explain the reason for permitting manual initiation of the RHR Service water pumps. In this regard, discuss the consequences if attempts were made to start these pumps during the emergency loading sequence.

5.7 Provide and justify the design bases for each of the rod block functions described on pages 7-3.2 through 7-3.4.

5.8 State and justify the design bases for the control rod position detection and indication (display) system(s). Provide a description and an evaluation to show that these bases have been satisfied. Since we anticipate that a requirement related to consecutive reed switch failures will be included in the

Technical Specifications, state the number of consecutive reed switch failures which can occur without restricting the use of the affected control rod system. If the number of consecutive reed switch failures is exceeded, will the hydraulic drive system for the affected control rod be valved off?

5.9 As stated in the FSAR, the instrumentation which initiates and controls the ECCS and containment isolation system is designed to meet the requirements of IEEE-279. However, the design bases for the reactor protection system listed on pages 7-7.1 and 7-7.2 do not include this requirement. Please resolve this apparent inconsistency.

5.10 Describe the design (physical and electrical) of the reactor protection system instruments located in the turbine building, and show that the design meets the requirements of IEEE-279.

5.11 Describe and evaluate the steam line high-flow instrumentation to show that it is separate and independent (physically and electrically) of like instruments which provide inputs to the feedwater controller. If the instruments are not separate and independent, your evaluation should show that no failure of the feedwater controller can preclude main steam line isolation.

5.12 Not all containment isolation signals are derived from four instrument channels with a trip logic of one-out-of-two twice. For each of those signals not so derived, state the number of channels being provided and their trip logic arrangement. Also, describe how each can be tested and calibrated during operation.

5.13 Describe and evaluate the ability to safely shut down the reactor should access to the control room be lost. This description should include, but not be limited to:

a. A discussion of remote (out of control room) instrumentation and controls presently available which would be needed to bring the plant to hot standby and maintain it at hot standby.

b. A discussion of the potential capability to bring the plant from hot standby to cold shutdown through the use of suitable emergency procedures.

5.14 In regard to the rod block monitor, the FSAR does not contain sufficient information related to the protection from single failures. Please identify in detail the areas of the rod block monitor system which lack redundancy or testability, and discuss the quality and equipment qualifications of the system components.

6.0 ELECTRICAL SYSTEMS

6.1 Detailed information on the offsite electrical system grid was presented in answers 4.0 and 7.0, Amendment 6 of the PSAR. However, it was noted on page 7.3 of Amendment 6 that ". . .not all additions or interconnections are necessarily good from the standpoint of system stability. For this reason, the networks must be continually re-examined to investigate the effects of changes in the network and changes in the load pattern." Therefore, update the evaluation presented in Amendment 6 where necessary.

6.2 From discussions with your representatives and from a review of a preliminary one-line diagram, we understand that you have modified the design of the onsite emergency power system. Your design now permits the automatic transfer of 480 vac motor control centers between redundant and otherwise independent onsite emergency power sources. Submit a description and an evaluation which justifies compromising the independence of these power sources by this automatic bus transfer feature.

6.3 There are two 348 kv and three 115 kv transmission lines emanating from the switchyard. State the design criteria for distributing these lines along rights-of-way. To show that the criteria are satisfied, describe the number of right-of-ways; the number, type, and size of towers per right-of-way; and the number of circuits per tower.

6.4 Submit a description and evaluation of (1) the switchyard circuit breaker controls and the power supplies for these controls, and (2) the automatic transfer from normal auxiliary to any of the reserve power sources. Your evaluation should show that no single failure can preclude the availability of offsite power to the engineered safety features. Also, state whether the switchyard controls are under the direct control of the reactor operator.

6.5 Since cooling of the diesel-generators is required in about one minute following actuation, explain why the initiation of the emergency service water pumps is not included in the diesel loading sequence presented in Table 8-4-1 (revised in Amendment 14).

6.6 Describe the preoperational tests and analysis which will be made to show that each diesel generator is capable of sustaining the loss of the largest load any time during a design basis accident or an emergency shutdown.

6.7 Describe and evaluate the consequences of single failures of the instrumentation and automatic relaying controls associated with the diesel generator emergency power system.

6.8 Provide an evaluation to show that electrical independence between redundant battery chargers and between diesel generators is not compromised by the circuitry which permits either battery charger to be energized from either diesel.

6.9 Evaluate the consequences of single failures on the instrumentation which automatically transfers to the redundant 125 vdc source upon failure of the normal source.

6.10 Provide the design and an evaluation of the instrumentation which monitors and controls the control room heating and ventilation system during or subsequent to a design basis accident.

6.11 The FSAR does not provide a description or evaluation of the instrumentation and controls of such vital auxiliary emergency systems as plant air system, RHR service water system, emergency service water system, reactor building cooling water system, etc. Accordingly, provide the design criteria, a description, and analysis of the design to support your criteria (concerned with only those portions of these systems which are necessary for safety).

6.12 Provide a listing of those safety related fluid systems which require heat tracing to assure continued availability. Also, provide the design criteria, and a description and evaluation of the designs to support your criteria.

6.13 Provide test data or equivalent information which will provide assurance that:

a. The seismic design requirements listed in the FSAR for the reactor protection system, containment isolation system, engineered safety feature instrumentation and control, and emergency power systems are satisfied.

b. The reactor level and pressure sensors can maintain the required accuracy and perform their design function during normal operation, expected transients and rapid depressurization (200 lbs/sec).

6.14 Please submit your cable installation design criteria for preserving the independence of redundant reactor protection system and engineered safety feature circuits (instrumentation, control, and power). For the purpose of cable installation, the protection system circuits should be interpreted in their broadest sense to include sensors, instrument cables, control cables, and power cables (both a.c. and d.c.), and the actuated devices (e.g., breakers, valves, pumps, etc.):

a. Cable separation should be considered in terms of space and/or physical barriers between redundant cables. Please address (1) the separation of power cables from those used for control and instrumentation, (2) the intermixing of control and instrument cables within a tray (or conduit, ladder, etc.), (3) the intermixing within a tray (or conduit, ladder, etc.) of cables for different protection channels, and (4) the intermixing of non-vital cabling with protection system cabling.

b. Discuss your criteria with respect to (1) the separation of penetration areas, (2) the grouping of penetrations in each area, and (3) the separation of penetrations which are mutually redundant.

c. Discuss cable tray loading, insulation, derating, and overload protection for the various categories of cables.

d. Discuss your criteria with respect to fire stops, protection of cables in hostile environments, temperature monitoring of cables, fire detection, cable and wireway markings, and the administrative responsibility for, and control over, all of the foregoing (a-d) during design and installation.

7.0 QUALITY ASSURANCE

7.1 Table C-4.1 in Appendix C of the FSAR lists various equipment vendors for items purchased by General Electric and Bechtel, and for which an NSP quality assurance team intends to make surveillance inspections at the particular shops. Have these inspections been completed? If not, when is the scheduled completion? Provide a summary of your findings, especially non-conformances and corrective actions.

7.2 Describe how the NSP quality assurance program would detect nonconforming material and prevent its being used.

7.3 Describe that portion of your quality assurance program which assures that adequate independent reviews or checks have been incorporated into the preparation of various procedures (e.g., pre-op testing, startup testing, operational procedures, maintenance procedures).

7.4 Numerous quality assurance documents are generated by various organizations throughout the construction of the facility. Describe your program which provides for the retention of these documents, including the types of documents to be retained, where they will be located, and the provisions provided for their maintenance.

8.0 GENERAL

8.1 At a meeting held on April 1, 1969, NSP representatives indicated that installation of a strong motion seismograph at the Monticello plant is being considered. Please confirm this, and indicate the number and type of instrument(s), and the placement of the instrument(s) in the facility.

8.2 Describe the provisions and/or special fuel handling procedures for protecting the spent fuel pool from loss of water from an accidental dropping of heavy objects, such as the fuel shipping cask.

8.3 To enable us to evaluate whether an operator would have adequate time to take appropriate and orderly action in an emergency, provide a detailed chronological listing of the operator actions that would be necessary following a design basis accident coupled with the simultaneous loss of offsite power to the facility.

8.4 A brief description on site access control is presented in Section 13.4.5 of the FSAR. However, the following additional information in regard to security measures is required:

a. Describe the security measures to guard against and to detect unauthorized access to the reactor site, control room, reactor building, and all other principal facility buildings.

b. Describe the extent of access to various portions of the facility (as indicated in (a) above) anticipated for the groups listed below. Also, describe the procedures to be used in admitting these groups:

(1) Members of the general public.

(2) Northern States Power Company employees stationed offsite; i.e., not part of the regular plant staff.

(3) Engineers, technicians, and contractors engaged by Northern States Power but not employed by Northern States Power.