Docket File

Docket No. 50-263

February 28, 1969

Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

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

Gentlemen:

In reviewing your application for the Monticello Nuclear Generating Station, we find that we need additional information to complete our evaluation to support issuance of a provisional operating license. The specific information required is listed in the enclosure. We anticipate that as our review continues, further information will be required.

Most of the requested information was discussed with your personnel at meetings held on January 23-24, 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.

In Section 6.0 of the attachment to this letter, we ask a number of questions concerning the medical support provisions for the Monticello facility. In this regard, you may be aware that the AEC conducts a series of training seminars for medical personnel who might be called upon in the event of a nuclear accident. We shall notify you of the scheduled dates of these seminars when that information is available. However, the assistance of our medical consultants will continue to be available in emergencies through our Regional Compliance Offices.

If you wish, you may respond by revising pages or sections to the FSAR, rather than by submitting answers to our questions as a separate supplement or amendment; however, if you choose the former, please provide cross references.

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

Sincerely,

/s/

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
List of Addl. Info. Required

Distribution:
AEC Pub Doc Rm
Docket File
DR Reading
RL Reading
RPB-1 Reading
C. K. Beck
M. M. Mann
P. A. Morris
F. Schroeder
R. S. Boyd
R. C. DeYoung (14)
L. Kornblith, CO (3)
D. R. Muller
N. Blunt
D. Vassallo
C. Hale

No RT input until 3-21.

OFFICE ▶	RL:RPB-1	RL:RPB-1	RL:RT	RL:RP	RL	RL
SURNAME ▶	Vassallo/eb	Muller	DeYoung	Boyd	Schroeder	Morris
DATE ▶	2/17/69	2/19/69	2/22/69	2/25/69	2/27/69	2/27/69

MONTICELLO NUCLEAR GENERATING PLANT

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

ADDITIONAL INFORMATION REQUIRED

1.0 REACTOR AND PRIMARY COOLANT SYSTEM

1.1 Quantitative leak detection is accomplished by monitoring the pumping rates of the equipment drain and the floor drain sump pumps. In this regard, provide the following:

a. The minimum detectable leak rate originating from any source, and the design bases supporting this leak rate.

b. The design bases to support your position for not including level indication in the sumps.

c. Details of the localized leak detection methods to be used were not presented in the FSAR. Provide a description, the number, and location of these devices. In addition, provide the bases to support the choice of methods employed.

1.2 The arrangement of the RHR system includes one supply line and two return lines to the reactor recirculation loops from the RHR heat exchangers. The design pressure of the supply line changes from 1250 psig to 150 psig outside the containment and after the two remote motor operated valves in series (valves #17 and #18). Provide additional documentation on how these values are instrumented to prevent their opening, or remaining open at pressures greater than the design of the low pressure portion of the supply line.

1.3 Evaluate the capability of the radiation monitor(s) associated with the main steam lines to promptly detect gross fuel failure. Provide the sensitivity and the bases for this sensitivity.

2.0 CONTAINMENT SYSTEMS

2.1 Provide the design bases of the filter-adsorber units in the standby gas treatment system with respect to filter-adsorber radioisotopic decay heat load. The design bases were not presented in the FSAR. Include supporting information to show that, in the event of a loss of one fan unit (subsequent to a design basis accident in which TID-14844 release fractions are assumed) the remaining fan unit has adequate capacity to cool the filter units in both parallel filter trains. Correlate this to the amount of radioisotopic material on the filters, the equivalent heat load, and the maximum temperature in the filters on both filter trains. Reference to previously conducted analyses would be acceptable, provided the applicability of the referenced analyses to the Monticello plant is clearly shown.

2.2 As stated on page 5-3.1 of the FSAR, blowout panels are provided to relieve reactor building pressure in excess of the design pressure (seven inches of water) in the event of a rupture of the primary piping within the building. In this regard, provide the following:

a. The design bases and description of the analysis used to size the blowout panel(s).

b. Describe how the blowout panel(s), function; i.e., how the reactor building is re-isolated following actuation of the blowout panel(s).

c. The offsite doses resulting from a primary system piping rupture (e.g., reactor cleanup system, RHR system) and actuation of the blowout panel(s), considering the elapsed time between initiation of release, detection, and final isolation. Describe the bases of the analysis, including sufficient information to support the choice of source term.

2.3 Upon detection of abnormal radiation levels within the reactor building, the normal ventilation system for the building is isolated and ventilation is automatically transferred to the standby gas treatment system. The FSAR does not specify the location of the radiation detector which initiates the automatic transfer, nor does it discuss how detector location affects the time to accomplish automatic isolation. Accordingly, provide the following:

a. The location of the radiation detector(s) and the elapsed time between initiation of release, detection, and final isolation or transfer of the reactor building ventilation to the standby gas treatment system.

b. To support your choice of radiation detector location(s), provide the results of an analysis which shows the offsite doses from a refueling accident, taking into account the elapsed time for isolation to be completed; i.e., the contribution of the total dose from building exfiltration. Include the assumptions and a description of the analytical method.

2.4 In evaluating the containment systems of all current generation water cooled power reactors, we have found that radiolytic decomposition of water following a loss of coolant accident may result in a flammable concentration of hydrogen and oxygen within the containment vessel. To evaluate this potential problem for the Monticello plant and complete our review of your application, the following additional information is required.

a. Provide a summary of the results of applicable analytical and experimental work completed to date on radiolytic decomposition of water, and indicate areas which are not yet complete.

b. Based on presently available or anticipated information, provide an evaluation of the safety significance of radiolysis products in the Monticello containment vessel after a loss of coolant accident. Include buildup of radiolysis products as a function of time, and the potential for and consequences of recombination.

c. Relate the potential water radiolysis problem to your position for not inerting the primary containment. Provide sufficient analytical evaluation to support your conclusion.

d. State the criterion which will be used to determine whether equipment to mitigate the presence of radiolysis products will be required. If such equipment is required, provide a design description, an evaluation of the adequacy of the design, and a description of applicable experimental work which would verify the design.

Reference to previously conducted studies or analyses would be acceptable, provided the applicability of the referenced analyses to Monticello is clearly shown.

3.0 <u>ENGINEERED SAFETY FEATURES</u>

3.1 At our meeting of January 23-24, 1969, we expressed concern regarding the protection of the ECCS pumps against flooding as a result of excessive leakage in the ECCS piping complex. Our concern is that, leakage significantly in excess of sump pump capacity might occur under post-accident conditions, and additional ECCS water inventory would be necessary to maintain long term emergency core cooling through the recirculation mode.

At the meeting NSP discussed its proposal to incorporate a crosstie to the RHR system which could maintain continuity of core cooling by supplying water to the core from the RHR service water pumps. However, because of the lack of isolation capability of the pump rooms, the continued addition of water through the RHR crosstie in the presence of a leak significantly in excess of sump pump capacity might eventually lead to flooding of the pump motors.

To evaluate your proposed design modification, the following additional information is required:

a. An assessment of the water depth in the lower level of the reactor building as related to the height of the ECCS pump motors, assuming the loss of all water from the suppression chamber.

b. The design description, design bases, and system logic for the proposed crosstie to the RHR system to maintain continuity of core cooling for leakage in excess of the sump pump capacity.

c. Assuming the requirement for long term operation of the crosstie resulting in leakage significantly in excess of sump pump capacity, describe what means will be provided to avoid possible complete flooding of the building.

3.2 Other plants which are similar in design to the Monticello facility have made a number of modifications to the system logic for the auto-relief valves. The modifications made include the following:

(1) The auto-relief valves are actuated coincident with initiation of the core spray system and LPCI system. This change was made to prevent any possible increase in pressure resulting from a reduction in depressurization effect from subcooled water flowing out a DBA break, such as discussed on page 6-2.18 of the Monticello FSAR.

(2) Automatic actuation of the auto-relief valves will be initiated by coincident indication of reactor low-low level and high drywell pressure. Previously, a third coincident signal indicating low flow in the HPCIS and/or feedwater systems was required to initiate actuation. This latter signal was removed, since it was recognized that a rupture in the HPCI or feedwater line upstream from the flow indicator could negate the auto-relief actuation signal.

(3) To provide a margin against valve failure, all valves above the minimum required for depressurization, are programmed to open upon receipt of the initiation signal.

(4) To prevent blowdown of the reactor vessel whenever a.c. power is not available and the auto-relief valves are signalled to actuate, an interlock is provided.

For each of the above items, discuss whether these design features will be incorporated into the Monticello facility. Provide appropriate sketches or logic diagrams to show how these design changes will be implemented.

3.3 Describe the pre-installation and post-installation testing program to be employed in assessing the performance of the auto-relief and safety valves. Discuss whether the reliability of the auto-relief valves can be compromised by entrained water in the valve internals.

3.4 In view of the problems that were encountered recently with plugging in the borating system of an operating nuclear plant, we have some concerns in this regard to the standby liquid control system for Monticello. At our meeting of January 23-24, 1969 your representatives indicated that means were being studied to maintain an adequate temperature at the discharge side of the borating storage tank. Describe the design and design bases for the system that will be used to maintain the temperature above saturation in the lines downstream from the storage tank. Also, describe the analysis and results to support the bases.

3.5 The capability of all instrumentation and electrical components which must function in the combined accident environment of temperature, pressure, and humidity should be verified through qualification testing. Accordingly, for the Monticello plant, identify all the above categorized components which: (1) have already been tested, (2) will require testing, or (3) have been purchased from manufacturers who have certified that suitable prior tests have been conducted.

In addition, describe the conditions under which the qualification tests have or will be conducted.

4.0 RADIOACTIVE WASTE CONTROL SYSTEMS

4.1 The main condenser air ejector off-gas subsystem, from the air ejector outlet to the stack inlet and including the 30-minute holdup line, is designed for a pressure of 350 psi. We understand that with this design, the air ejector subsystem should be capable of containing a possible explosion resulting from the hydrogen and oxygen normally present in the air ejector effluent (expected composition -- about 50% hydrogen, 25% oxygen, 10% air, and 15% water vapor).

To assist in the evaluation of this subsystem, the following additional information is required.

a. Provide by reference or analysis, the bases for the 350 psi design pressure of the air ejector subsystem.

b. Since the off-gas, high efficiency, particulate filters most likely would not survive a hydrogen explosion, calculate the radiological consequences. Consider the filters to have accumulated the maximum amount of activation products; i.e., for the expected end of filter-life. Also, provide sufficient information to support your choice of the source term for accumulated products on the filters.

c. All subsystems of the off-gas system, including the standby gas treatment system, terminate at the base of the plant stack. Because of this interconnection, evaluate whether a hydrogen explosion in the air ejector subsystem could create sufficient back pressure to cause any damage to the dilution fans and standby gas treatment system.

5.0 PLANT AUXILIARY SYSTEMS

5.1 The intake structure is the source for the circulating water system. In addition, it provides water for various engineered safety features. It is, therefore, imperative that this water source be available at all times. Provisions are made for de-icing during power operation, but the capability for de-icing following an accident was not addressed in the FSAR. Accordingly, provide additional details on the design bases, and operating and post-accident modes of operation for the intake structure de-icing system.

5.2 Identify those areas in the facility which have automatic sensing devices for fire detection and describe the devices employed. Also, identify those areas which will be provided with automatic fire extinguishing coverage and describe the equipment to be used. Since the fire protection water supply distribution system is not designed to Class I, describe how the disabling of this system, in the event of an earthquake, could affect the automatic fire extinguishing equipment and plant fire fighting capability in general.

6.0 CONDUCT OF OPERATIONS

6.1 At a meeting with NSP representatives, we discussed in some detail the need for amplification and clarification of a number of areas pertaining to the Plant Operations section of the FSAR. These areas where additional information is required are noted below.

6.1.1 Provide an organizational chart that designates the line authority and relationship between NSP, GE, and Bechtel before and after plant acceptance by NSP.

6.1.2 Define the duties, functions, responsibilities, and authority of the operating staff.

6.1.3 In regards to the functioning of both the Operations Committee and the Safety Audit Committee prior to and following acceptance of the plant from General Electric, clarify what constitutes a quorum and the anticipated frequency of holding meetings.

6.1.4 Discuss the background and experience of personnel to be used in a consulting capacity to the Safety Audit Committee.

6.1.5 Describe the technical support functions to be provided by the NSP engineering personnel as backup to the operations staff.

6.1.6 Outline the operating staff positions at Monticello and specify the minimum qualifications required of personnel to fill these positions.

6.1.7 To demonstrate the relationship between the training schedule and the plant startup schedule, including staff participation during this phase, provide a bar-type graph, and additional details on the training program.

6.1.8 Describe the retraining program planned for onsite personnel as well as for replacement personnel. Describe the refresher courses planned, and the frequency or scheduling anticipated for the courses.

6.1.9 Discuss the actual participation of the staff in the formulation and writing of the various plant procedures.

6.1.10 More details should be included in the Pre-operational and Startup Test Procedures sections of the FSAR pertaining to:

a. Scope.

b. Objectives.

c. Scheduling of tests (i.e., % power, etc.).

d. Prerequisites.

e. Design criteria (basis and verification).

f. Acceptable deviation between operating and test conditions, including basis for the deviation.

g. Evaluation of the proposed tests.

h. Evaluation of the test results.

i. Individual responsibilities and authority before, during, and after the actual tests are conducted.

6.1.11 Describe the types, nature of, and frequency of reports NSP plans on submitting to the various AEC Regulatory Divisions.

6.1.12 Provide more comprehensive information on the Emergency Plan for the Monticello facility.

6.2 For some time we have been reviewing the status of medical plans, including availability of facilities and personnel at or near licensed facilities to provide for care of contaminated or irradiated persons in case of a radiation emergency. This is a continuing study and our objective is the development of a rule covering these matters. In the meantime, in order to assist us in this study and in the consideration of your application for a provisional operating license, provide the following information.

6.2.1 A description of onsite facilities for decontamination and immediate emergency treatment of injured personnel, including details concerning the following:

a. Decontamination space including size, location in reference to plant population and operations and/or hazards relating to radioactivity, shielding provided, shower and water availability, waste control and disposal.

b. Existing emergency medical care facility, including location, equipment, supplies, plumbing, waste disposal and hours of availability on round-the-clock basis.

c. Equipment and supplies available for immediate gross decontamination of personnel, including injured.

d. Equipment, supplies, written procedures and standing orders for immediate control and emergency treatment of injured personnel.

6.2.2 A description of qualifications, professional education and special training (e.g., training in supervision and care of injured radioactively contaminated persons and other occupational health hazards) of:

a. Resident professional personnel, such as physicians, nurses, industrial hygienists, and health physicists.

b. Resident semi-professional personnel, such as nursing assistants, and health physics technicians.

c. Readily available offsite professional medical personnel.

d. Provisions for meeting cost of services rendered by each of the above persons.

e. Hours of onsite duty, offsite availability, location and distance from site, and hospital staff appointments currently held for personnel listed in (a) through (c) of 6.2.2.

6.2.3 A description of arrangements for transport of injured personnel, including:

a. Equipment and supplies for in-transit emergency treatment.

b. Standing orders for emergency procedures kept in vehicles.

c. Location of vehicles, including distance from site and average time to respond to call.

d. Availability on round-the-clock basis.

6.2.4 Identification and location (including distance from site) of hospital agreeable to accepting:

a. Patients for further decontamination.

b. Contaminated injured personnel, for treatment including:

(1) Description and location of special facilities designated for contaminated patients.

(2) Description and location of special facilities for treating radiation injuries.

(3) Equipment, including surgical facilities and supplies for handling radiation or contamination victims.

6.2.5 Qualifications of professional medical personnel at the support facility (hospital or clinic) to treat radiation and contamination victims, including number and types of physicians and a description of any specialized training related to contamination or radiation injuries.

6.2.6 A description of any limitations that exist regarding availability of offsite medical facilities and support, with particular regard to:

a. Time of admission of accident casualties.

b. Length of stay for contaminated patients.

c. Extent of contamination or direct radiation levels associated with injuries.

d. Types of injuries or illnesses.

e. Any special limitations on admission or treatment, such as indemnification of the medical facility by the licensee.

6.2.7 Presence of written plan and standing orders in receiving area of hospital detailing actions to be taken and procedure to be followed when contaminated person with or without injury is brought to hospital.

If plan and orders are not posted, is hospital willing to:

a. Have such plan and orders readily available for emergency use?

b. Instruct professional and administrative staff about plan and orders?

7.0 SAFETY ANALYSIS

7.1 Provide the assumptions, description and results of an analysis to show that a massive failure of the turbine-generator will not result in missiles which can damage the control room or other vital safety features of the plant. Reference to a previously conducted analysis or analyses would be acceptable provided sufficient information is supplied to clearly show the applicability of the referenced analysis to the Monticello plant.