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IN REPLY REFER TO:

Docket No. 50-263

DEC 2 7 1966

Northern States Power Company
414 Nicollet Avenue
Minneapolis, Minnesota 55401

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

Gentlemen:

On October 11-12, 1966, and again on December 2 and 12, 1966, representatives of Northern States Power Company met with the staff of the Division of Reactor Licensing to discuss your application for a Construction Permit and Facility License for a nuclear power plant at a site near Monticello, Minnesota.

During the earlier meeting, it became evident that changes to the engineered safeguards, as described in Facility Description and Safety Analysis Report, were being developed. To assure an accurate understanding of the proposed engineered safeguards, as well as related systems and analyses, please provide answers to the questions listed in the attachment. The staff will be available to discuss and amplify the meaning of any of these questions, should this be necessary.

A supplement to the original application on the subject of field erection of the reactor pressure vessel was recently submitted and is currently being reviewed. Specific staff questions on the supplement are not included in this letter, however we request that as a further supplement to information on field erection, you provide a comprehensive history and evaluation of operating experience (at least U.S. experience) with field-erected pressure vessels. Please emphasize high-temperature high-pressure vessels.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
List of questions

DRL	DRL	DRL
RSBoyd/SJ	EGCase	PAMorris
12/27/66	12/27/66	12/27/66

REQUEST NO. 1 FOR INFORMATION ON

NORTHERN STATES POWER COMPANY MONTICELLO UNIT NO. 1

1.0 Accident Analysis

Sections 5.1.3.1, 5.1.3.2, 13.3.4.2, and 13.3.4.4 of the Preliminary Safety and Analysis Report should be supplemented. The following is required:

1.1 Provide a technical description of the analytical blowdown model with an assessment of the adequacy of the model.

1.2 Provide an evaluation of the blowdown forces on the reactor vessel internal structures including thermal transients which would occur during a primary coolant blowdown or post MCA condition. Discuss the potential forces and structural capability for the various internals considered.

1.3 Provice a quantitative evaluation of the blowdown forces generated in the upper reactor vessel region following maximum steam line rupture, i.e., on separators, baffles, driers, etc.

1.4 Provide an integrated presentation of the system behavior following the MCA (double-ended recirculation line break) with particular attention paid to the following aspects, including appropriate assumptions, justifications, identifications of operating modes and the particularly sensitive error bands on the calculated values.

Reactor Internal Pressure Differences

Reactor Coolant Water Levels

Containment Thermodynamic Response

Core Fuel Thermal Characteristics (including percent fission gas released from fuel as a function of fuel temperature and time)

1.5 Discuss the accident recovery conditions from the MCA (no core cooling case) for the situation in which the containment spray coolers were inoperative, particular attention being paid to the pressure and temperature characteristics of the containment.

1.6 Provide an evaluation of the behavior of the control rods and the fuel bundles for a spectrum of depressurization events. Attention should be paid to the ability of the control rods to be inserted and other possible reactivity effects.

1.7 Provide a discussion of the design margins (with respect to delayed cooling effects) involved with the provided engineered safeguards of core flooding and core spray with respect to the MCA.

1.8 Provide the following information with respect to the liquid boron injection system:

Minimum boron concentration in the liquid poison and in the total primary system volume as well as a reactivity balance for the spectrum of reactor conditions under which liquid poison should insure reactor shutdown. Include coolant temperature, coolant void, pressure, reactivity and nuclear power as a function of time.

1.9 Provide an analysis of the loss-of-coolant accident where the core remains covered with water and radiolytic decomposition of the water occurs. Include an analysis of the potential hydrogen and oxygen buildup and a discussion of any hazard potential that might result. This analysis should include justification of the g-values used.

1.10 Discuss the damage level and design margins of the primary system and internals with respect to reactivity transients as influenced by the release of molten fuel into the coolant. Discuss the importance of void collapse reactivity effects due to pressure pulse and capability of nuclear overpower circuits to respond to rapid power bursts and cause the control rods to scram. What is the effect of fuel burnup on these analyses?

2.0 <u>Engineered Safeguards</u>

2.1 Discuss the differences in concept and degree of redundancy, if any, between the safeguards for this plant and those proposed for the Quad-Cities reactor.

2.2 Provide the design basis for each of the safeguards systems including flow rates, system descriptions, heat transfer capacities, initiating signals and instrumentation logic.

2.3 Provide the design basis for the standby diesel-generator including a system description, accident power loads, initiating signals, instrumentation logic, reliability and accessibility during accidents. Show typical pump motor starting loads vs. time. Justification for not providing redundant on-site power capability (e.g., a second diesel-generator) should be included in this discussion.

2.4 Provide a description of instrumentation used to monitor post accident recovery to assure control of release of radioactivity to the environs.

2.5 Identify the range of primary system breaks for which each of the engineered safeguards systems will provide adequate core cooling.

2.6 Discuss the capability of the automatic-depressurization system and the conditions which require its use.

2.7 Provide the design basis for and the reliability requirements for the main steam isolation valves. Discuss prototype tests to demonstrate closure capability during accident conditions and leak tightness during post MCA recovery.

2.8 Clarify the following:

2.8.1 On page II-6-4 a statement is made about using the spectra of Figure II-6-5 and damping values from Table II-6-3. Later in the same paragraph of Section II-6-3.1 the statement is made, "if computerized methods of dynamic analysis are used, the mathematical model may be subjected to an excursion through the Taft earthquake of July 21, 1952 North 69 West component with an applied factor of 0.33." The statement then goes on to indicate that the structure should be examined under values of twice those given in Fig. II-6-5 as well, or a dynamic excursion with an applied factor of 0.66. Clarify the meaning of this factor in terms of its use in the procedure, and whether the maximum earthquake corresponds to values twice those indicated in the spectra of Fig. II-6-5.

2.8.2 On page II-6-5 it is noted that for Type 2 structures and equipment a minimum seismic horizontal coefficient of 0.10 with a one-third allowable increase in basic stress will be used in the design. State the reason for selecting this value and its consistency as compared to the procedures adopted for the Type I structures. Provide the basis of the response acceleration spectrum of Fig. II-6-5. In order that we may analyze more readily the short period range of this spectrum, please provide this portion of the spectrum on an expanded scale or provide a logarithmic plot of the spectrum. How is the uncertainty considered in calculation of period using a response spectrum showing a large change in acceleration response for a small change in period?

2.8.3 Damping values are listed in Table II-6-3 on page II-6-5. The damping level for reinforced concrete structures is listed as 5% critical. We believe that the damping value is a function of the stress level permitted either under design conditions or for safe shutdown. What is the basis for the 5% figure?

2.9 What provisions are taken to insure stability of cranes during an earthquake?

2.10 Provide the design basis, reference design, structural integrity analysis and proposed surveillance program for the facility stack.

2.11 Some clarification of the treatment of Class I equipment contained within Class II structures is requested. Are there any such items required for safe shutdown? If so, how is the analysis handled for these items and how is their response effected by the response of the Class II structures to which they may be attached?

2.12 Provide a description of the inspection procedures to be followed during the construction of the containment and other critical structures, and identify the organizational responsibility for quality control and inspection.

2.13 Provide a description of the design considerations which reflect the requirement that the containment wall accommodate the stresses and deformations which might be imposed by earthquake loads or pipe breaks without impairing containment integrity.

2.14 Describe the design criteria for the biological shield around the drywell.

2.15 Describe the design basis for protection of the containment and engineered safeguards against internal and external missiles.

2.16 Provide the design basis and functional description for the reactor building to torus vacuum breakers.

2.17 Provide a discussion comparing the containment design bases for the Monticello facility to those used for Dresden 2 and 3 and Quad-Cities 1 and 2 facilities. Include such considerations as pressures, free volumes, energy inputs (containment capability) from blowdown and metal-water reactions, and typical drywell and torus wall thicknesses.

2.18 Provide the criteria for reactor vessel and coolant system inspection and describe how these criteria will be satisfied. In particular, identify those areas of the vessel which can be inspected and outline tentatively a program to be followed over vessel lifetime. Discuss the requirements for hydrostatic tests above design or operating pressure and any plans to revalidate system integrity over vessel lifetime. Is there any reason why such tests could not be performed?

3.0 Site Analysis

3.1 Provide a statement and evaluation, based on the 1980 population projections, of the low population distance for this site.

3.2 Provide a description and evaluation of any anticipated meteorological program to be initiated at the Monticello site.

3.3 Provide an analysis showing the minimum dilution to be expected between the condenser discharge outfall and the intakes for the nearest public drinking water supplies for both an accidental slug release and continuous release of radioactive effluent associated with (1) normal, and (2) low flow in the river. State the maximum amount of liquid radioactive waste which will be stored in the various on-site containers in relation to the capacity of these same containers and the maximum rate at which these amounts could be released.

3.4 Provide data on storage capacity and an estimate of the length of time withdrawal of drinking water can be suspended for the municipal water supplies down river from Monticello. Provide data on the water storage capacity and the rate of consumption of population centers within 50 miles downstream of the site.

3.5 Describe the cooling tower complex and its operation. Also discuss any nuclear safety considerations associated with the operation of the cooling towers.

3.6 Provide the justification for the estimated liquid radioactivity discharge rates of 1 mc/day normally and 250 mc/day with fuel leaks. Please discuss liquid effluent discharge in relation to 10 CFR 20 limits.

3.7 Describe the control room ventilation system and criteria for design.

3.8 Provide the justification for the exfiltration rates in high winds described in Section 5.2.2.2. Describe procedures and test frequency to periodically demonstrate leak tightness of the reactor building.

3.9 Describe the design bases for the plant to withstand the wind and pressure effects of tornadoes.

3.10 Provide design criteria and a detailed description of the reactor building standby gas treatment system.

4.0 <u>Power Plant</u>

4.1 Discuss the increase in the boron concentration (5000 ppm vs. 3800 ppm for Quad-Cities) in the temporary control curtains. What experience is related to this change? Discuss the structural integrity of the curtains as a function of the proposed irradiation period.

4.2 Describe the work which assures that cavitation in the jet pumps will cause no adverse vibrational effects during normal or abnormal operation such as loss of feedwater.

4.3 Describe the procedures used to determine the individual engineered hot channel factors. What is the accuracy or error band? How were these values used in the core thermal performance calculations?

4.4 Section II.2.2 of the Facility Description and Safety Analysis Report states that the main condenser will accommodate a 15% load rejection. Section 11.2.3 recognizes that the bypass valve will pass up to 15% of the throttle steam directly to the main condenser, and that the combined capacity of the bypass valves (three 5% capacity valves) and relief valves is sufficient to keep the reactor safety valves from opening in the event of a sudden loss of full load on the turbine generator. The increase in neutron flux, however, according to Section 11.3.1 causes a reactor scram. With reference to the foregoing what is the maximum step change in load that can be tolerated without scram, and what effect does the 15% bypass capability have? Describe reactor power level, pressure, recirculation flow, bypass steam flow, as a function of time after step power reductions.

4.5 Identify all components (e.g., power sources, transmission lines, protection devices, protective relaying and communication systems) which are pertinent to maintaining auxiliary site power following an abrupt shutdown of the Monticello plant from 100% power. Describe the expected behavior of the NSP power network and relate this behavior to the reliability of site power. List the minimum out of service equipment, failures, and mal-operations which will result in the loss of normal auxiliary site power following this transient.

4.6 With regard to the core thermal analysis and considering the effect of flow distributions produced by orificing, what will be the maximum exit quality in the hottest channel? Identify the MCHFR of 1.5 with regard to quality, flow rate, and location in the core. What confidence is there that this ratio will not be lessened by axial power distributions other than the reference distribution considered in your analysis? If the MCHFR of 1.5 is reached, what will be the increase in power necessary to reach the critical heat flux?

4.7 Provide an analysis of the buildup of tritium in the primary coolant over the life of the plant. Consider such sources of tritium as diffusion of fission product tritium through the cladding, activation of additives or impurities in the primary coolant, if any, neutron reactions with boron and photonuclear reactions. Evaluate the hazard from tritium inventory in the primary coolant in terms of a steam line rupture. What means of tritium monitoring will be provided to ensure that excessive concentrations are not reached in the reactor coolant or radwaste system?

4.8 State the excess reactivity for the hot critical and hot full power (free and equilibrium xenon) condition. Show net reactivity, xenon reactivity coolant temperature and power level as a function of time if, as a result of equipment failures, only half of the withdrawn control rods scram.

4.9 What is the maximum allowable reactor vessel coolsown and heatup rates for a normal reactor shutdown and startup, respectively?

4.10 What instrumentation senses a steam line break and causes the steam line isolation valves to close?

4.11 It is stated that the plant may be controlled either by recirculation flow control or control rod movement. Please describe the relationships and interdependency of these controls.

4.12 Discuss possible courses of action in the event of individual LPRM (in-core neutron sensors) failures to assure that effective core protection is maintained.

4.13 How is accidental control rod movement out of the approved operating pattern detected automatically and what corrective action occurs? For example, if a control rod drifts or unintentionally moves when other selected rods are actuated while in automatic mode of operation and is undetected by the operator, what would prevent automatic features from compensating the local disturbance and thus propagating or extending an unevaluated operating condition?

4.14 Describe the reactor safety considerations associated with fire in the control room.

4.15 State the criteria for detection of primary system leakage within the drywell and the bases for corrective action. Describe the methods and instrumentation that will be used.

4.16 Provide a comparative analysis of the natural circulation capability of the Monticello reactor and the Oyster Creek (or Nine Mile Point) reactor to support the following statement: "The higher natural circulation potential of the jet pumps system also tends to improve the 'heat-flux-coolant flow' relationship which results from a complete loss of power to the recirculation pumps" (pp. IV-2-3 FD&SAR).