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Docket No. 50-263

February 27, 1967

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Northern States Power Company
414 Nicollet Avenue
Minneapolis, Minnesota 55401

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

Gentlemen:

To verify the understanding reached during our February 2 and February 24, 1967 meetings, enclosed is the list of additional information which we stated was necessary to complete your application.

Your reply to these questions should be submitted as an amendment to your application.

Sincerely yours,

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
List of Questions

RPB #1/DRL	DRL	DRL
DRMuller/emb	RSBoyd	PAMorris
2/24/67	2/24/67	2/25/67

ADDITIONAL INFORMATION REQUIRED

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 Based on the review of Amendment No. 2 to the FDSAR, Northern States Power Company Monticello Nuclear Generating Plant "Design, Fabrication and Erection of the Reactor Vessel," and in order to obtain further assurance of the integrity of the field fabricated reactor vessel, we request that the following information be provided:

1.1 Typical vessel test plate metallurgical evaluations, including macroetch cross sections showing the full thickness of the base plate and the weld metal and a thorough exploration of the heat affected zones by hardness surveys. In addition, the results of typical metallographic examinations of base metal, weld metal, and heat affected zones should be provided for correlation with mechanical properties. As an alternative, you may wish to provide the results of the tests outlined above on the Monticello vessel material.

1.2 From each vessel test plate, including typical plates used for welding qualification, sufficient specimens should be obtained so that the complete Charpy temperature transition curve can be established for the base metal, heat affected zone, and weld metal. The radius of the Charpy specimen notch should be checked using a magnifying radius comparator. The other sensitive dimensions should also be measured. The root of the notch should be located as close to the coarse grain region as possible (as determined by etching) and should be wholly within the heat affected zone, with the notch oriented parallel to the plate thickness. Please describe your plans concerning the foregoing.

1.3 A chemical analysis of each plate, forging, and vessel test plate weld should be performed which includes the trace elements, such as soluble and total Al, Ti, Sb, Sn, Pb, Cu, V, Zr, etc. Please describe your plans concerning this analysis.

1.4 The outer surface of the reactor vessel should be examined by magnetic particle and ultrasonic tests after all post weld heat treatment and the 125% overpressure test have been completed. The purpose of the examination is to detect any cracks caused by the heat treatment or the overpressure test. Please describe your plans concerning such a test.

2.0 We have observed that core flooding after an MCA may cause a reactivity transient if control rods fail to scram in part or in toto. As a means of providing an understanding of the magnitude of this occurrence, discuss the consequences of potential transients that might occur should all or various numbers of control rods fail to scram after an MCA. Discuss the basis upon which such an occurrence is considered to be not credible. Assuming a number of control rods fail to scram, discuss possible means of limiting such transients.

3.0 We note that the turbine bypass capability, 15%, is less than the 25, 40, and 100% provided on other BWR power plants. Your response in Amendment 4 to question No. 4.4 provides the calculated response of the plant to typical transients expected during plant operation. In order to obtain a clear understanding of the safety significance of this lower bypass capability, please provide calculations similar to those of Figures 4.4-1 through 4.4-5, assuming a step load rejection from rated power and assuming a 100%, 40%, 25%, and 15% turbine bypass capability. How would the number of reactor scrams be affected by the bypass capability? Discuss your reasons for selecting 15% rather than a higher bypass capability.

4.0 From the information submitted to date, it is evident that proper functioning of the engineered safeguards is predicated on their starting almost immediately after the MCA. Since this can best be assured by providing the most reliable source of off-site power possible, it would be desirable to demonstrate that the effects of a loss of the Monticello generator on the power network are in accordance with predicted performance characteristics. Please provide your comments and outline any plans you may have formulated to perform such a test. The benefit and independence of the six transmission lines should also be explained.

5.0 The inerting provisions in the pressure suppression containment protect against the combustion of hydrogen which might be generated in metal-water reactions after a loss-of-coolant accident. The radiolytic decomposition of water, however, could result in oxygen as well as hydrogen evolution. Your consideration of this problem in question 1.9 in Amendment No. 4 assumed that the decomposition would be equivalent to that observed during normal reactor operation. Our calculations, based on experimental data at one atmosphere, indicate that this may be a potential long-term problem. In our calculations, the yield of hydrogen and oxygen was calculated using the constant: G = 0.45 molecules of hydrogen per 100 ev of gamma energy absorbed in the water. (Reference: "Reaction Mechanism Leading to the Formation of Molecular Hydrogen in the Radiation Chemistry of Water" by E. Haydon and M. Moreau, J. Phy. Chem. Dec., 1965). Please report on your further consideration of this potential problem including whether additives are available which would retard the reaction.

6.0 Your response to question No. 1.7 in Amendment 4 discussed the conservatism in post MCA recovery calculations. However, our interest is related to the consequences of spraying cold water on the core or flooding the reactor vessel after core temperatures have risen above 3000°F if for any reason the engineered safeguards fail to respond in the precise manner and time assumed in the calculations we have reviewed to date. Accordingly, please describe the consequences of core cooling delays beyond those which assume the diesel starts automatically within a few seconds after MCA. Is it correct to assume that full core spray and core flooding, regardless of the delay, are desirable, or is there some reason for altering post MCA recovery procedures if spray and flooding were delayed?

7.0 As noted during recent telephone conversations with your staff, our criterion concerning emergency power for engineered safeguards is that in the event of loss of all off-site power, sufficient sources of alternate power shall be provided to assure a capability for performing the functions required of the engineered safeguards. For Monticello, we interpret this to mean that in the absence of other sources of on-site power, at least two diesel generators should be provided. Each should have the capability outlined in your response to question 2.3 in Amendment 4.

8.0 Geologic and Seismic Design

8.1 Please provide the locations of the principal structures, including turbine and reactor buildings, intake structure, stack and diesel building. Provide foundation elevations, foundation soil strata, soil bearing capacities and the loads to be imposed by the respective buildings. Indicate which structures will be separate due to seismic response considerations. The drawings should indicate the location of borings in the immediate vicinity of the structures, and these boring logs should be included.

8.2 Provide a description of the design of the intake structure including foundation and seismic considerations.

8.3 Provide the justification for not performing dynamic tests on the foundation soil beneath the reactor building to determine whether differential settlement could take place. Discuss potential liquifaction of the soil beneath the reactor building.

8.4 Are interaction loadings between the reactor building substructure and the surrounding soil considered in the design of the building substructure for both static and dynamic loading conditions?

8.5 Please provide the reasoning for the selection of the Taft, rather than the El Centro, earthquake spectrum. A comparison of both the computed spectra and the averaged spectra should be presented for the Taft and El Centro earthquakes. Discuss the agreement of the time-history record used in the computer analysis with the response spectrum throughout the frequency range.

8.6 Please verify that the stresses arising from the earthquake in both the vertical and horizontal direction, and which occur simultaneously at a particular location, will be added directly to the stresses arising from the other applicable loadings, including pressures and temperature stresses arising from an accident.

8.7 A table of damping coefficients is given on page II-6-5. It is noted therein that for the "reactor-building (massive construction with many cross walls and equipment and providing only secondary containment)" a damping value of 5 percent is specified. Further elaboration on this point is given in answer to question 2.8 of Amendment 4. As a result of Dr. Newmark's recent considerations, he would be in agreement with this value for cases in which working stresses are no more than about one-half the yield point and in which there may be considerable cracking associated with the concrete structure. In the event that the concrete is not stressed to that level where it is considerably cracked, he recommends a value of 2 or 3 percent as being more reasonable. Please discuss this recommendation in view of your proposed design.

8.8 We understand that 5 percent rather than 10 percent critical damping for ground rocking modes of vibration will be used in this design. Please confirm this understanding.

8.9 Verify that the damping factors cited in Table II-6-3 are to be employed for both design and maximum earthquake loading conditions.

8.10 We understand that a minimum seismic coefficient of 0.05 rather than 0.10 will be used for Class II structures. Please show that the proposed value is conservative.

8.11 Dr. Newmark has recommended a damping value of 2 or 3 percent be used in the design of the stack. In view of this recommendation, please discuss the damping values proposed for this design.

8.12 Please provide details concerning strengthening of areas around penetrations of the containment, particularly the drywell, to insure that the required strength and ductility under earthquake and service loading are attained.

8.13 Please clarify the comment relating to safe shutdown of the plant in tables on pages V-3-2 and V-3-3 of FDSAR, Vol. I. It is not clear whether special stress criteria will be employed for Loading Condition 2 or 3 respectively (for safe shutdown) or whether these references apply to the stresses listed previously in the tables.

8.14 Table V-3-3 (FDSAR) refers in Loading Condition 3 to a factor of "M.O.L. + 2 x S.L." This factor of "2 x S.L." should be changed to reflect the maximum earthquake that is selected, which may not necessarily be twice the design earthquake nor twice the response values applicable thereto.