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WButler

Docket No. 50-263

MAY 19 1977

Northern States Power Company
ATTN: Mr. L. O. Mayer, Manager
Nuclear Support Services
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Gentlemen:

By letter dated November 22, 1976, you requested our approval of a modification to provide redundant lifting features for the Monticello Nuclear Generating Plant Reactor Building Crane.

We have completed our review and have determined that the proposed modification is acceptable, subject to the following conditions:

(1) The carry height of the IF-300 70 ton cask shall be administratively limited to a maximum of the minimum height necessary to gain floor clearance during cask swing, plus two (2) inches; and

(2) The carry height of the NFS-4 and NAC-1 casks, approved for use in our January 25, 1977 letter, shall be limited to a maximum of six (6) inches; and

(3) The travel path of all spent fuel shipping casks shall be within the limits established in your January 22, 1976 and June 16, 1976 submittals; and

(4) Loads of weight greater than 1 fuel element (excluding the crane load blocks and associated tackle) shall not be transported directly over spent fuel stored in the spent fuel pool without prior NRC approval.

We request that you provide verification/commitment to the above items within 30 days of receipt of this letter.
0
Our related Safety Evaluation is enclosed.

Sincerely,
Original signed by
Don K. Davis
Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

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DATE ➤	5/12/77	5/12/77	5/12/77	5/13/77	5/19/77	

Enclosure:
Safety Evaluation

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSIO
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING APPROVAL OF CRANE MODIFICATIONS AND USE OF

70 TON SPENT FUEL SHIPPING CASK IF-300

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

Introduction

In its January 22, 1976 submittal, the Northern States Power Company (NSP) proposed an interim program which would use the existing Monticello overhead crane handling system for offsite shipment of spent fuel. Further, NSP stated that the proposed interim program would be replaced with a permanent cask handling program once the long term program had been completed and implemented.

With certain qualifications, e.g., limiting cask weight to 25 tons, the NRC, by letter dated January 25, 1977, concluded that the proposed interim cask handling program was acceptable.

On November 22, 1976, NSP completed its study of the permanent cask handling program and submitted a description of its proposed crane modifications and a safety analysis of the proposed 85 ton (rated load) reactor building crane system. In addition, on February 28, 1977, NSP responded to an NRC request for certain additional information, regarding the proposed reactor building crane system. NSP proposed to handle the 70 ton IF-300 spent fuel shipping cask with the modified reactor building overhead crane.

Discussion and Evaluation

The reactor building overhead crane system is required for handling heavy loads during refueling operations and during operations involving the offsite shipment of spent fuel. The heaviest load that has to be handled during refueling operations is comparable to the 70 ton IF-300 spent fuel shipping cask. However, this load is only handled when the plant is in a cold shutdown condition. Further, NSP states that sufficient diversity in equipment exists to maintain the reactor in a cold shutdown condition should any one of the refueling loads be dropped. Therefore, this operation does not pose a significant safety hazard.

An analysis previously submitted by NSP indicates that the plant's structures are not capable of withstanding the drop of a 70 ton shipping cask. However, the interim program for offsite shipment of spent fuel was limited to the use of a 25-ton cask. The travel path of the cask between the transporter, laydown area, and the spent fuel pool cask loading area was established by the licensee. This travel path passes over those portions of the structure most capable of withstanding a cask drop accident. If the carrying height of the cask above the operating floor does not exceed six inches, the structures are capable of withstanding the drop of a 25-ton cask as indicated by NSP analyses. Figure 3-1 (attached) shows the travel path of the cask in relation to the supporting members below the operating floor.

In its November 22, 1976 submittal, NSP proposed to upgrade the overhead crane system by making certain modifications. These modifications will consist of replacing the existing trolley and hoisting system. Within the constraints of available space and requirements relative to performance, movement and weight, the new trolley and redesigned hoisting system will, where practical, provide a dual load path, single-failure-proof hoisting system which complies with the provisions of draft NRC Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants." The non-redundant components, e.g., main hook, load block, shafts and structures will have increased load safety factors to reduce the likelihood of their failure.

A new trolley will be required to accommodate the new main hoist within the existing space, clearance and safety requirements without restricting its lifting capacity of travel envelope.

The double drum main hoist will use a unique quad-support reeving system. Two redundant parallel spur gearing and speed reducers will be provided to deliver power to both ends of the dual main hoist cable drums. Each power train has been designed for the duty cycle and projected life of the crane. Both gear trains will be mechanically connected to one another via the main hoist motor shafting. A failure in either of the two power trains would not result in loss of control of load. Following a postulated failure and the removal of any obstructions resulting from the failure, the system would still be capable of performing the lowering and hoisting functions.

A drum retention structure will be incorporated with each of the drums to provide backup support in the event of a postulated drum shaft, bearing or machinery support failure.

The number of main hoist holding brakes has been increased from two to three. Each brake is designed to hold 125 percent of the hoist motor torque at base speed. The brakes will be set upon: (1) loss

of electrical power to the brakes; (2) reduction of the hoist motor voltage to 70 volts; (3) the actuation of either the upper or lower hoist travel limit switches; or (4) the loss of power to the main hoist motor. The circuitry has been arranged such that the brakes will be sequentially applied with a 0.5 second delay interval for the second and third brakes.

The reeving system consists of four 6 x 37 EIPS-IWR 7/8 inch diameter wire ropes. The proposed system has a minimum factor of safety of 5:1 under rated load. Due to reeving and equalizer systems, should any one rope fail the load will be maintained in a safe, stable condition by the remaining three wire ropes.

The reeving is such that one end of each of the four rope sections terminates at one of the two load equalizer floating pistons. The two equalizers are double acting hydraulic cylinders which equalize the cable loads and compensate for normal rope stretch by slow movement of the floating hydraulic pistons. Movement of the floating piston causes hydraulic fluid to flow from one end of the cylinder to the other end through a velocity limiting device. In the event of a postulated failure of one rope, the velocity of the hydraulic fluid being moved from one end of the hydraulic cylinder to the other side creates a large pressure drop. Therefore, the hydraulic cylinder will act as a dashpot which reduces the shock of transferring its load to the remaining intact ropes. Switches have been provided to detect abnormal displacement of the floating pistons. When these switches are actuated, the hoist system will be deactivated and the holding brakes set.

Due to the existing plant structural limitations and lift height requirements, the maximum wire rope interior fleet angle will be two degrees twenty minutes rather than the one and one-half degrees recommended in Regulatory Guide 1.104, but is less than the two degrees thirty minutes allowed in AISE Standard No. 6, "Specification for Electric Overhead Traveling Cranes for Steel Mill Service." Further, to assure the integrity of the rope NSP will perform rope inspection, replacement, and maintenance in accordance with ANSI B 30.2-1967, "Safety Code for Overhead and Gantry Cranes". At the rated load of 85 tons, the non-redundant main hoist hook, load block trunnion, hook nut, hook thrust bearing, upper sheave nest supoort structure and load blocks will have a factor of safety of 10:1.

Moreover, "two blocking" redundant hoist limit switches, actuated by the load block, will be provided to preclude excessive cable loads from being developed due to the lower load block contacting the upper load block. To avoid a "load hangup" while lifting any critical load

in the equipment hatch, the bridge and trolley will be properly positioned for the lift and then power will be locked out on these drives during the time of the lift. Further, additional protection against overload conditions has been provided by incorporating two overload sensing devices on each half of the redundant reeving system. These load cells will trip the main hoist motor and set the holding brakes should the load reach 125 percent of design rated load.

To prevent overspeed, an overspeed switch will be incorporated in the existing main hoist control system which will trip the hoist drive motor and set the holding brakes.

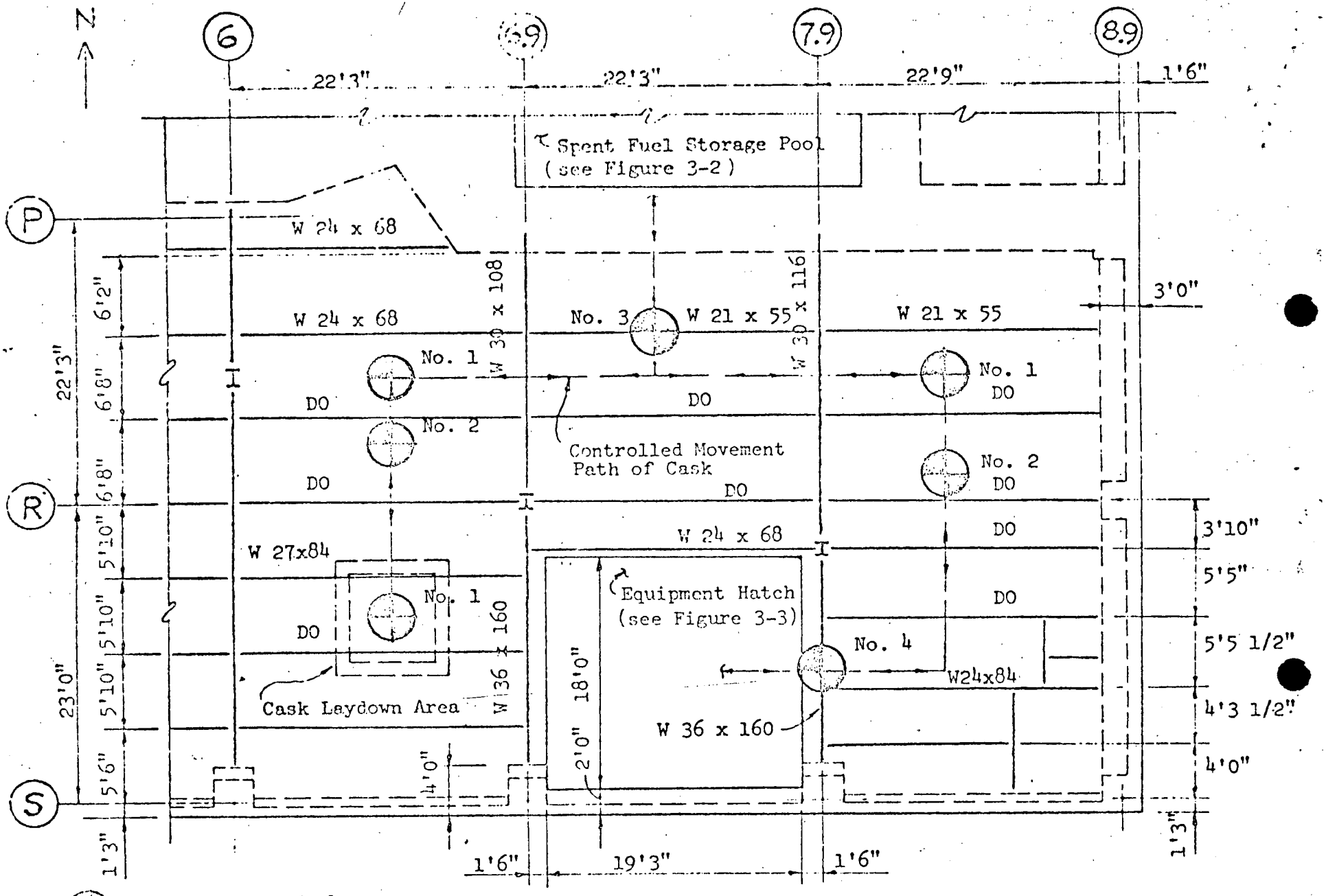
Inspection and testing will be carried out on a periodic basis as required by ANSI B30.2.0 and OSHA Part 1910, Section 1910.179.

To provide further assurance that a postulated cask drop accident will not occur when handling the IF-300 cask, NSP states they will only use the redundant IF-300 cask handling yoke.

Conclusion

We find that NSP's proposed modifications to the reactor building crane has incorporated all the provisions of draft Regulatory Guide 1.104 that are practical for the Monticello design. We conclude that in addition to the proposed modifications to the reactor building crane, the licensee has proposed adequate measures to preclude the occurrence of a cask drop accident and to mitigate its effect in the very unlikely event that it should occur. Therefore, the proposed permanent modifications to the reactor building overhead crane are acceptable.

Date: MAY 19 1977



Postulated cask drop location with bottom end impact

STRUCTURAL FRAMING PLAN
FOR THE OPERATING FLOOR AT ELEVATION 1027'-8"

FIGURE 3-1

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MAY 19 1977

Docket No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer, Manager
Nuclear Support Services
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Gentlemen:

RE: MONTICELLO NUCLEAR GENERATING PLANT UNIT 1

The purpose of this letter is to advise you that, as a result of our continuing review of information related to the Mark I Containment Program, the NRC staff has revised its previously expressed position regarding the acceptance criteria for removal (or reduction below 1.0 psid) of required drywell-wetwell differential pressure controls. Our current position is described in Enclosure 1 and should be considered prior to any request for authorization to remove or reduce differential pressure control requirements.

In addition, as discussed at the February 4, 1977 meeting between the NRC staff and representatives of the Mark I Owners Group, we have reassessed our position regarding utilization of the test data from the NRC-sponsored 1/5th scale testing program currently in progress at Lawrence Livermore Laboratory. Our current position is described in Enclosure 2 and is provided for your information.

If you have any questions regarding this information we would be pleased to discuss them with you.

Sincerely,

(S)

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Acceptance Criteria for Removal or Reduction of Drywell-Wetwell Differential Pressure Controls
2. Application of Data from the Lawrence Livermore Laboratory 1/5th Dynamics Test Program

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SURNAME	RPSnaider:esp	JGuibert	DKDavis			
DATE	5/17/77	5/18/77	5/19/77			

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ENCLOSURE 1

ACCEPTANCE CRITERIA FOR THE REMOVAL OR REDUCTION OF DRYWELL - WETWELL DIFFERENTIAL PRESSURE CONTROLS

The loading criteria for the Short Term Program's (STP) plant unique analyses utilized the base case downward loads taken from the 1/12 scale Phase II (December) test series. This was found acceptable primarily because the Phase II tests represented a larger data base for the base case (no differential pressure) condition and because there was reason to believe that the downward load anomaly observed in the Phase III (January) test results was caused by facility configurational problems. Additional consideration was given to the load sensitivity curves for differential pressure control (ΔP); which were developed using Phase III test data. The ΔP load sensitivity curves account for a fraction of the downward load anomaly, depending on the magnitude of the differential pressure.

In meetings with the Mark I Owners Group during February 2-3, 1977, some preliminary results from the 1/12 scale Phase IV tests were presented. The purpose of this test series was to investigate the cause of the downward load anomaly observed in the Phase III tests. The preliminary results of the Phase IV tests, while showing an influence of the natural frequency of the test facility, tend to confirm the higher magnitude of the downward loads observed during the Phase III tests.

Therefore, for those plants whose licensees propose to operate without differential pressure controls, we will require that the licensee determine the effect of a 33% increase[1] in the downward load, and subsequently demonstrate a limiting stress ratio of less than 0.5 (factor of safety greater than two) for the critical structural element, consistent with the STP requirements for "most probable load". In making this evaluation, we will find acceptable the assumption of a linear relationship between the downward load and the stress ratio. Further, for those plants whose licensees propose to reduce the magnitude of the differential pressure, because of the normalization of the Phase III data to the Phase II downward load, operation of ΔP control below 1.0 psid will not be allowed.

This position has been developed to allow the removal of the differential pressure control requirements with an adequate margin of safety to permit the continued investigation and resolution of the downward load anomaly. Once the downward load anomaly has been resolved, we will apppropriately revise the criteria for the removal or reduction in differential pressure controls.

(1) NEDC 20989 P (Addendum 2), Loads and their Application for Torus Support System Evaluation, page 105.

ENCLOSURE 2

APPLICATION OF DATA FROM THE LAWRENCE LIVERMORE LABORATORY POOL DYNAMICS TEST PROGRAM

During meetings with the Mark I Owners Group on February 2-3, 1977, we discussed use of the forthcoming data from the Lawrence Livermore Laboratory (LLL) pool dynamics test program in conjunction with the Long-Term Program (LTP). As you know, the NRC has undertaken the test program at LLL to provide confirmatory hydrodynamic load data for the Mark I configuration.

Based on our review of the Mark I owners revised Program Action Plan, we have found that the current test programs have several deficiencies relating to three-dimensional pool swell effects. We believe that these deficiencies will result in an NRC requirement for additional margins to account for the associated uncertainty, prior to its application in the LTP.

The LLL test facility, on the other hand, does not have these deficiencies, and will provide confirmatory data useful in the further resolution of three-dimensional pool swell loads for the Mark I containment design. We, therefore, recommend that the Mark I owners make provisions in the LTP to utilize the data from the LLL air test series for the purpose of confirming the method (analytical or empirical) that will be used to establish the hydrodynamic pool swell loads.

Provisions have been made to have the Mark I owners represented during our discussions on the LLL test programs and to provide the data obtained from the program to the Owners Group on a timely basis. The Mark I owners should be in a position to use the data from the LLL program just as they would data from any other source.

Central Files

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

May 16, 1977

Mr. Ernst Volgenau
Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Volgenau:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

As requested in your letter of April 15, 1977, copies of the notice enclosed with your letter have been posted.

Yours very truly,

L. J. Wachter

L. J. Wachter
Vice President - Power Production
and System Operation

cc: Mr. James G. Keppler
Mr. Victor Stello
Mr. G. Charnoff
Minnesota Pollution Control Agency
Attention: Mr. J. W. Ferman

9