The basis for this conclusion was presented in our Safety Evaluation dated May 21, 1976, a copy of which is enclosed. Our initial review of your August 29, 1977 submittal confirms this evaluation.

Sincerely,

Original Signed by
V. Stello

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation
dated May 21, 1976

cc w/enclosure:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE➤	DOR:ORB#2	DOR:ORB#2	OELD	DOR:AD/OR	DOR:AD/OT	DOR:DIR
SURNAME➤	MGrotenhuis*	DKDavis*	MGrossman*	KRGoller*	DGEisenhut*	VStello
DATE➤	12/ 1 /77	12/1/77	12/1/77	12/1/77	12/1/77	12/ /77

Northern States Power Company -2- DEC 2 1977

The basis for this conclusion was presented in our Safety Evaluation dated May 21, 1976, a copy of which is enclosed. Our recent review of your August 29, 1977 submittal confirms this evaluation.

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Victor Stello, Jr., Director
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Office of Nuclear Reactor Regualtion

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SURNAME➤	MGrotenhuis:ak	DKDavis	Krossman	KGoller KRG	DEisenhut	VStello
DATE➤	12/1/77	12/1/77	12/2/77	12/1/77	12/2/77	12/ /77

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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:

By letter dated August 29, 1977, you submitted a document entitled "Removal of Drywell-Wetwell Differential Pressure Controls" for the Monticello Nuclear Generating Plant. You stated that in an August 23, 1977 meeting, the NRC staff had informed you that, as prerequisites to removal of the differential pressure control, you must either demonstrate compliance with the criteria of the NRC's May 19, 1977 letter ~~or show~~ that torus loads are less than allowable by code. The drywell-wetwell differential pressure control had been established in accordance with the NRC letter dated February 27, 1976, and was committed to in your letter of March 1, 1976.

We have reviewed your August 29, 1977 submittal and have concluded that the drywell-wetwell differential pressure controls can be removed at the Monticello Nuclear Generating Plant. However, we have also concluded that future technical specifications for wetwell water level instrumentation will be necessary to assure that the level envelope of the Monticello Mark I Containment Short Term Program Plant Unique Analysis will remain valid. This issue will be the subject of future discussions with your staff.

Our safety evaluation related to the elimination of the drywell-wetwell differential pressure control is enclosed.

Sincerely,

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

Enclosure:
Safety Evaluation

WB 11/4/77
BLB 11/4/77

OELD
JScinto
(S H Lewis) 11/14/77

cc w/enclosure: See next page

OFFICE	ORB#2:DOR	PS/OT:DOR	PM/OT:DOR	EB/OT:DOR	AD/OT:DOR	ORB#2:DOR
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DATE	11/ 4 /77	11/ 4 /77	11-6-77	11/10/77	11/10/77	11/ /77

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 2, 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

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The NRC staff still has the matter of long term removal of Δp controls under review. We anticipate that the review should be completed in the near future. In telephone conversations with the NRC staff, you have indicated your intent to conduct, on behalf of the Mark I owners group, further in-plant safety/relief valve tests at Monticello in the near future. On the basis of information presented in your letter dated May 10, 1976 and on the further information in your August 29, 1977, letter regarding the in-plant safety/relief valve tests we have concluded that the requirement for maintaining a 1 PSI drywell-torus differential pressure during performance of in-plant safety /relief valve testing may be relaxed provided:

1. The differential pressure is restored during any unscheduled interruption of the testing which is anticipated to exceed 24 hours.

2. The differential pressure is restored within 24 hours after a test is completed.

The basis for this conclusion was presented in our Safety Evaluation dated May 21, 1976, a copy of which is enclosed. Our initial review of your August 29, 1977 submittal confirms this evaluation.

Sincerely,

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation
dated May 21, 1976

cc w/enclosure:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING TEMPORARY RELAXATION OF THE DRYWELL-TORUS DIFFERENTIAL PRESSURE REQUIREMENT AT MONTICELLO NUCLEAR GENERATING PLANT FOR TESTING PURPOSES

DOCKET NO. 50-263

INTRODUCTION

By letter dated May 10, 1976, Northern States Power Company (NSP) submitted plans for the performance of an In-Plant Safety/Relief Valve Test at the Monticello Nuclear Generating Plant during the week of May 23, 1976. To provide meaningful results, such testing will require an "unbiased" initial water level in the relief valve blowdown line and thus the temporary (approximately 96 hours) cessation of the 1 PSI drywell-torus differential pressure voluntarily imposed by NSP after the February 26, 1976 meeting of the BWR Mark I owner's group and the NRC staff.

BACKGROUND

A January 6, 1976 letter from Mr. Ivan F. Stuart of General Electric to Mr. R. S. Boyd of the USNRC described an In-Plant Safety/Relief Valve Test to be performed as part of the long-term program for evaluation of Mark I containment systems. NSP agreed to perform the test at the Monticello Nuclear Generating Plant. Testing was originally to have been performed in March 1976. However, reanalysis of the torus support downward load/strength ratios in February 1976 indicated that the margins of safety of the torus structure were not as large as had been previously calculated. Thus, at the meeting of February 26, 1976, NSP, along with other Mark I licensees, agreed to establish the differential pressure between the drywell and the torus to provide a reduction in the potential loads during a postulated loss of coolant accident and an associated restoration of the margins of safety to obtain a factor of safety of about two. This agreement was confirmed in a February 27, 1976 letter from Mr. Benard C. Rusche, Director, Office of Nuclear Reactor Regulation, NRC, to NSP. Subsequently, NSP has undertaken a program of structural modifications to enhance the safety margins of the Monticello torus. A description of the modifications and the resulting loading tables are included as an attachment to NSP's May 10, 1976 letter.

DISCUSSION AND EVALUATION

The NRC staff agrees with NSP and the Mark I owner's group that the In-Plant Safety/Relief Valve Test will provide data essential for the development of long-term solutions to the Mark I problem. Our concern is mainly that there is reasonable assurance that the torus would remain intact and function as intended should the extremely low probability Loss-of-Coolant Accident (LOCA) occur during the relatively short period of time (96 hours) that the differential pressure would be relaxed. In their May 10, 1976 letter, NSP committed to restore the differential pressure should the test be interrupted for periods anticipated to exceed 24 hours.

NSP has submitted data, as an attachment to its May 10, 1976 letter, which supports the NSP conclusion that the modifications, which have been completed, produce a 46% improvement in ratios of load to ultimate capacity without the differential pressure. Improvement of over 100% is claimed for up-loads. We have completed a preliminary review of data which has been provided and agree with NSP's statement concerning the increased safety margins with the modifications installed and the differential pressure removed. Additionally, we concur that conditions under which the test will be performed "will provide safety margins conservatively within and consistent with the margins of safety discussed in Mr. Rusche's letter of February 27, 1976" (i.e., factors of safety of about two).

The test does not involve an unreviewed safety question in that (1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased, (2) the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created, and (3) the margins of safety as defined in the bases for the technical specifications are not reduced.

On the basis of the above considerations, we have concluded that elimination of the drywell-torus pressure differential during the relatively short period of time required to perform the In-Plant Safety/Relief Valve Test is acceptable.

Date: May 21, 1976