

FROM: Northern States Power Company
 Minneapolis, Minnesota 55401
 Arthur V. Dienhart

DATE OF DOCUMENT: 7-30-70	DATE RECEIVED: 7-31-70	NO.: 2399
LTR: x notarized 7-30-70	MEMO:	PORT:
OTHER:		

TO:

 Dr. Peter A. Morris

ORIG.:	CC:	OTHER:
1 signed & 37 conf'd		
ACTION NECESSARY <input type="checkbox"/>	CONCURRENCE <input type="checkbox"/>	DATE ANSWERED:
NO ACTION NECESSARY <input type="checkbox"/>	COMMENT <input type="checkbox"/>	BY:

CLASSIF: U
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 REG. NO:

FILE CODE: 50-263 *Applic.*

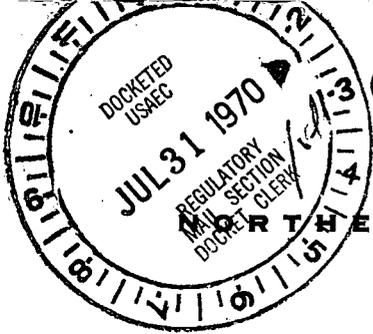
DESCRIPTION: (Must Be Unclassified)
 Ltr submitted in re to our 6-26-70 ltr
 furnishing comments on ACRS ltr of
 6-15-70....

REFERRED TO	DATE	RECEIVED BY	DATE
Muller W/4 cys for ACTION	7-31-70		
DISTRIBUTION:		DO NOT REMOVE	
Regulatory file			
AEC PDR			
Compliance (2)			
OGC(Rm P 506A)			
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Maccary		D. Thompson	
Dromerick (3)		DTLE(Laughlin)	
Holt		NSIC(Buchanan)	
Minogue		Case(Reading File)	
Moore		ACKNOWLEDGED	

ENCLOSURES: 

REMARKS:
 NOTE: Messenger advises that 2 addl
 signed & notarized cys will be mailed.
 Holding (12) cys

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NSP

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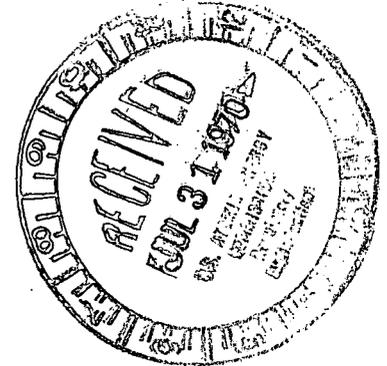
NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

Regulatory

File Cy.

July 30, 1970



Dr. Peter A. Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D.C. 20545

Dear Dr. Morris:

MONTICELLO NUCLEAR GENERATING PLANT E-5979
Comments on ACRS Letter of June 15, 1970

This is in reply to your letter of June 26 requesting NSP to submit our responses to each of the comments cited in the ACRS letter of June 15, 1970. The following paragraphs constitute our response to the ACRS letter items.

1. Independent Stress Analysis Check - Work is proceeding on an independent review of the thermal stresses in the as-built piping within the drywell including the recirculating system and associated portions of the residual heat removal system. Included will be a review of the piping layouts, design assumptions, computer input data, and calculational methods. In addition, one or two stress analysis computations will be verified by performing a complete and independent stress computation. This check review will be performed by groups in the design organizations different from the groups that conducted the original designs. The work is scheduled for completion in the fall of 1970, and the results will be summarized in a report, copies of which will be furnished to you.

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2. Hot Displacement Measurements - The movement of piping systems within the drywell resulting from thermal expansion during system heat-up will be evaluated by the following actions:
- a) Record the as-built locations and ambient temperature settings for pipe hangers on pipe in the drywell over two inches in diameter.
 - b) Record accessible hanger positions at one intermediate temperature (300°F to 400°F) for piping larger than two inches in diameter on the recirculating, feedwater, steam, core spray, control rod drive return, and high pressure coolant injection systems in the drywell.
 - c) Inspection will be made at the intermediate temperature to visually verify that adequate piping clearance exist and freedom of movement through penetrations is possible.
 - d) Extrapolation to operating temperature of the results and observations made in Item b.
 - e) Adjustment of hangers, if necessary, to bring supports within specifications.
 - f) Where possible observe and record displacements at full operating temperature.
 - g) Recheck cold hanger settings on accessible hangers at the completion of the first total thermal cycle of the plant.

3. High Points in Non-Flowing Parts of Primary System Piping

A study has been initiated of non-flowing parts in the primary system piping aimed at disclosing pockets where gases could become trapped. Systems involved in this study include the

core spray, recirculation, standby liquid control, control rod drive, head spray and vent, jet pump instrument, and other miscellaneous instrument lines. Based on the study work completed to date, certain corrective measures have been taken in the form of added vents. The study is continuing and completion is scheduled for the fall of 1970, after which a report will be produced and copies of this report will be furnished to you.

4. Biological Shield Study - A study is now underway to determine the effects of a safe-end to nozzle weld failure on the biological shield surrounding the reactor vessel. Completion of this study is scheduled for the fall of 1970, and the report will be produced and copies furnished to you. Any plant modifications required as a result of the study will be made at the first scheduled refueling outage.
5. In-Service Inspection - Your staff has discussed with us a schedule of inspections for safe-ends and other plant components involved by our modification program related to sensitized stainless steel. The attached table outlines the additions to our in-service inspection program appropriate to our modification program.
6. Leak Detection - The performance of the reactor coolant leakage detection system, including the speed and sensitivity for detection, will be evaluated during the first eighteen months of plant operation and the conclusions of this evaluation will be reported to the AEC. Modifications to the leak detection system,

if any are required, will be performed during the first scheduled refueling outage following the evaluation. In addition, we will continue to study other techniques for detecting leaks and examine techniques for appropriate disposition in regard to the Monticello facility.

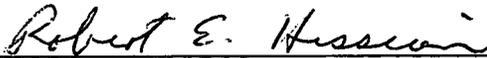
Yours very truly,



Arthur V. Dienhart
Vice President - Engineering

Subscribed and sworn to before me

This 30 day of July, 1970



Robert E. Hessian
Notary Public, Hennepin County, Minnesota.

My Commission Expires May 15, 1976

(Notarial Seal)

SUPPLEMENTAL IN-SERVICE INSPECTION PROGRAM

<u>Examination Area</u>	<u>Examination Method</u>	<u>Inspection Interval</u>
Field-clad safe-ends	PT; UT	One safe-end at or within first refueling outage Other safe-end at or within second refueling outage
Welds of field-replaced safe-ends	PT; UT/RT	Two safe-ends at or within first refueling outage Two safe-ends at or within the second refueling outage
Severely weld-sensitized heat-affected zones in wrought piping (Type 304, 316)	PT; UT/RT	10% at or within first refueling outage
Field clad-repaired sensitized components within reactor vessel whose failure could adversely affect safety	Visual	Examination to the extent practicable at the first and second refueling outage, and whenever the reactor vessel head is removed for other purposes, within the first and second refueling

PT - Liquid Penetrant Examination

RT - Radiographic Examination

UT - Ultrasonic Examination