NRC FORM 195 (2-76) U.S. NUCLEAR REGULATORY COMMISSION

# NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

| | |
|---|---|
| DOCKET NUMBER | 50-263 |
| FILE NUMBER | |
| DATE OF DOCUMENT | 2-27-76 |
| DATE RECEIVED | 3-1-76 |
| NUMBER OF COPIES RECEIVED | 39 |

TO: MR D L ZIEMANN

FROM: NORTHERN STATES POWER CO
MINNEAPOLIS, MINN
L O MAYER

☐ LETTER ☐ NOTORIZED ☒ UNCLASSIFIED ☒ COPY

PROP | INPUT FORM

DESCRIPTION

LTR, RE OUR 2-4-76 LTR... .L FURN RESPONSES TO QUESTIONS ON MSL SETPOINTS AND MCPR W/ ATCH DRAWING.... ....

ENCLOSURE

ACKNOWLEDGED
DO NOT REMOVE

PLANT NAME: Monticello

SAFETY — FOR ACTION/INFORMATION — ENVIRO 3-4-76 rkb

| | | | | |
|---|---|---|---|---|
| | ASSIGNED AD : | | | ASSIGNED AD : |
| ✓ | BRANCH CHIEF : (6) | ZIEMANN | | BRANCH CHIEF : |
| | PROJECT MANAGER: | | | PROJECT MANAGER : |
| ✓ | LIC. ASST. : | Diggs | | LIC. ASST. : |

INTERNAL DISTRIBUTION

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| ✓ | REG FILE | | SYSTEMS SAFETY | | PLANT SYSTEMS | | ENVIRO TECH |
| ✓ | NRC PDR | | HEINEMAN | | TEDESCO | | ERNST |
| ✓ | I & E (2) | | SCHROEDER | | BENAROYA | | BALLARD |
| ✓ | OELD | | | | LAINAS | | SPANGLER |
| | GOSSICK & STAFF | | ENGINEERING | | IPPOLITO | | |
| | MIPC | | MACCARY | | | | SITE TECH |
| | CASE | | KNIGHT | | OPERATING REACTORS | | GAMMILL |
| | HANAUER | | SIHWEIL | | STELLO | | STEPP |
| | HARLESS | | PAWLICKI | | | | HULMAN |
| | | | | | OPERATING TECH | | |
| | PROJECT MANAGEMENT | | REACTOR SAFETY | ✓ | EISENHUT (LTR) | | SITE ANALYSIS |
| | BOYD | | ROSS | ✓ | SHAO | | VOLLMER |
| | P. COLLINS | | NOVAK | ✓ | BAER | | BUNCH |
| | HOUSTON | | ROSZTOCZY | ✓ | SCHWENCER | | J. COLLINS |
| | PETERSON | | CHECK | ✓ | GRIMES | | KREGER |
| | MELTZ | | | | | | |
| | HELTEMES | | AT & I | | SITE SAFETY & ENVIRO | | |
| | SKOVHOLT | | SALTZMAN | | ANALYSIS | | |
| | | | RUTBERG | | DENTON & MULLER | | |

EXTERNAL DISTRIBUTION

| | | | |
|---|---|---|---|
| ✓ | LPDR: Minneapolis, MN | NATL LAB | BROOKHAVEN NATL LAB |
| ✓ | TIC | REG. V-IE | ULRIKSON(ORNL) |
| ✓ | NSIC | LA PDR | |
| | ASLB | CONSULTANTS | |
| | ACRS 16 HOLDING/SENT | | |

CONTROL NUMBER

1923

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401



February 27, 1976



Mr. D. L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ziemann:

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

Response to 2/4/76 Questions on MSL Setpoints and MCPR

Your February 4, 1976 letter requested additional information on our December 1, 1975 request for changes to the Technical Specifications. Questions 1 and 2 deal with the proposed reduction of the main steamline low pressure setpoint which is a generic matter. Questions 3 and 4 deal with proposed changes to Monticello minimum critical power ratio (MCPR) limitations. The latter is a more urgent concern in that a delay in implementing these changes needlessly threatens full operating capacity of the plant. Should your review of the main steamline low pressure setpoint change require more time than that of the new MCPR limits, we request that the two issues be separated and the MCPR changes be issued as soon as possible. The questions and their respective answers are as follows:

NRC Request # 1

For the spectrum of steamline breaks downstream of the main steamline isolation valves (MSIV) provide the following:

(a) An analysis of the change in the radiological consequences resulting from the reduction in the setpoint for MSIV closure on low steamline pressure from 850 psig to 825 psig. So that we may perform an independent check, also provide the difference in the amount of steam and liquid released as a result of the lower setpoint.

(b) A discussion of the effects of the setpoint reduction on peak cladding temperature and MCPR.

Response # 1

The accident postulated does not rely on the main steamline low pressure setpoint to initiate an isolation and scram. The main steamline flow

1923

sensors provide such protection. A September 17, 1975 letter from Mr. L. O. Mayer (NSP) to Mr. R. S. Boyd (USNRC) entitled, "Main Steam Line Flow Trip Setting" analyzes the spectrum of break sizes and shows the radiological consequences to be well within 10 CFR Part 100 limits. The proposed setpoint change in no way affects the reported radiological consequences of accidents involving steamline breaks.

The effects of the setpoint reduction on peak cladding temperature and MCPR are discussed in response to question 2 below.

NRC Request # 2

> In the analysis of the failure of the turbine pressure regulator presented in your SAR, the main steamline isolation valves are assumed to start closing (initiating a reactor scram) when the low steamline pressure is reached.
>
> (a) Identify other transients that assume MSIV closure and reactor scram are initiated by the low steamline pressure signal.
>
> (b) Provide a reanalysis of the failure of the turbine pressure regulator transient, and other transients identified in (a), assuming MSIV closure and reactor scram at the proposed setpoint of 825 psig.

Response # 2

The main steamline low pressure sensors were installed to provide reactor isolation for the abnormal operational transient associated with failure of the initial turbine pressure regulator in the open direction. No credit is taken for the sensors in any of the other analyzed abnormal operational transients or postulated accidents.

The present isolation setpoint, 850 psi, was selected quite arbitrarily. The transient analysis presented in the FSAR shows the turbine pressure regulator failure to be a very insignificant event. Being familiar with the progression of minor reactor dynamic perturbations, one can conclude with confidence that there would be no significant changes if the isolation setpoint were at 825 psi. The initial intent of our submittal was to support the change qualitatively without the plant-specific analysis so as to avoid taxing industry expertise with trivial calculations. Since you have requested such an analysis, we would like to reference a bounding analysis done for the Hatch I unit, Docket Number 50-321, submitted October 9, 1975 by Mr. Chas Whitmer of Georgia Power Company. The Hatch analysis shows that a main steamline low pressure setpoint change from 880 to 825 psi involved no significant changes in the transient results. The increase in pressure

along with a flow decrease results in essentially no change in MCPR. Because of the similarities between Hatch and Monticello, and the fact that Monticello is requesting a smaller setpoint change (850 to 825 psi) than analyzed for Hatch, the Monticello transient results are expected to be even less significant. Being such a mild transient, the peak cladding temperature is of less concern than that of bounding transients such as a turbine trip without bypass which is routinely analyzed. Also, the parameters which affect cladding temperature might be studied from the Monticello FSAR, Figure 14-5-7. Failure of the initial pressure regulator in the open direction decreases pressure which causes greater moderator voiding, resulting in a rapid decrease in neutron flux which occurs essentially simultaneously with a scram. During this time core flow gradually decreases to approximately half of its initial condition. The removal of the heat source with continuous cooling results in a reduction of cladding temperature throughout the transient.

NRC Request # 3

Were MCPR values of 1.38 and 1.29 for 8x8 and 7x7 fuel used as the initial thermal conditions for establishing the worst case for rod withdrawal error? If so, what is the rod block setting and do the affected fuel bundles stay above a MCPR value of 1.06?

Response # 3

The rod withdrawal error was analyzed using the assumptions discussed in topical report NEDO-20360, "GE/BWR Generic Reload Licensing Application for 8x8 Fuel", Revision 1, Supplement 2, May, 1975. One of these assumptions is that the maximum worth rod is fully inserted and adjacent rods are withdrawn in a manner which will allow full design reactor power with operating limits attained near the inserted rod. In the case of the Monticello Reload-4 analysis, the fuel was assumed operating at the MCPR limits of 1.38 for 8x8 fuel and 1.29 for 7x7 fuel. The rod block monitor (RBM) setpoint was assumed to be 108%. It was found that even if the operator ignores all alarms during the course of this transient, the RBM will stop rod withdrawal while the critical power ratio (CPR) is still greater than the 1.06 MCPR safety limit.

NRC Request # 4

Provide the scram reactivity curve for EOC5.

Response # 4

The attached figure shows the scram reactivity used in the Cycle 5 analyses. This is conservatively derated to 80% of the expected value.

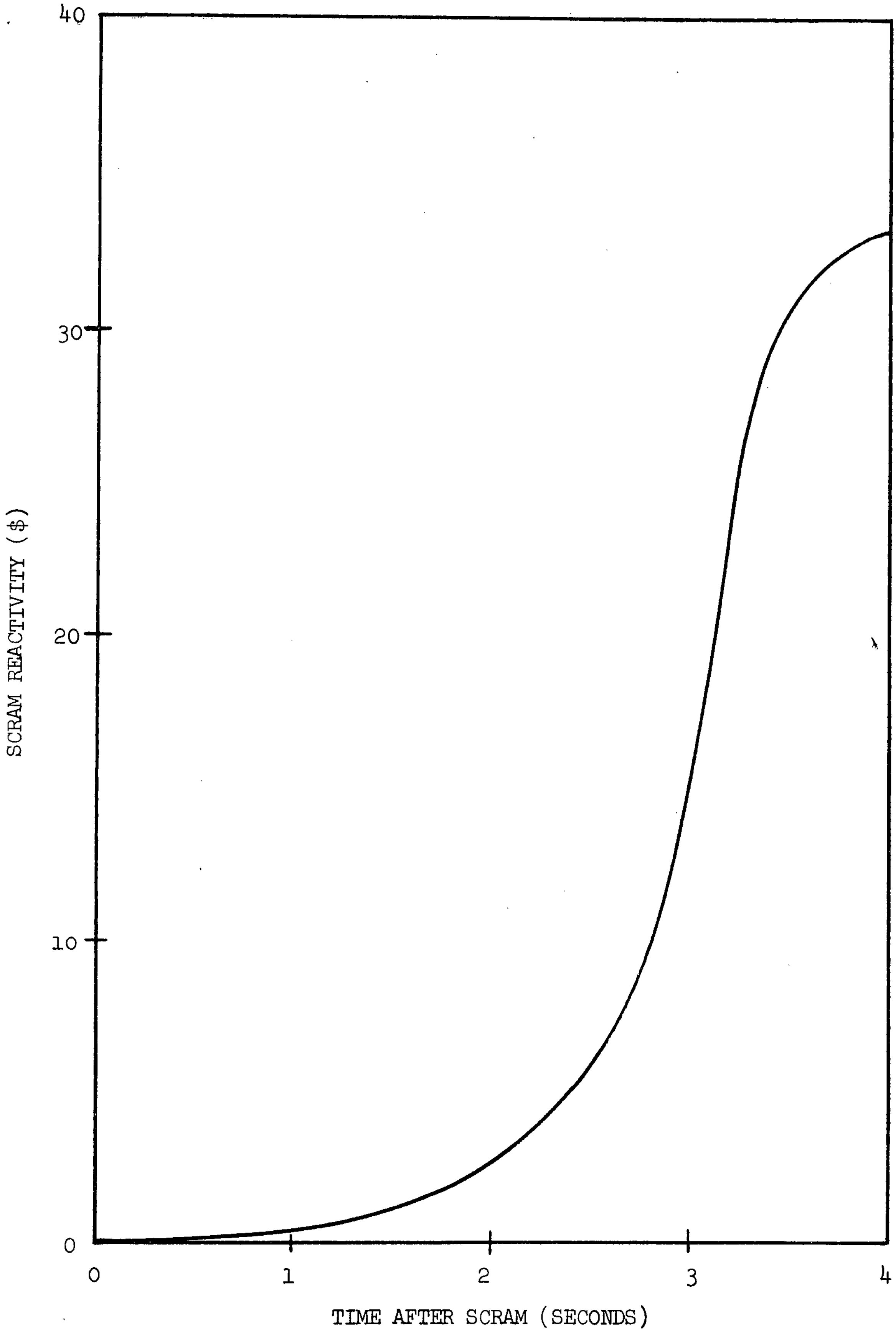
Yours very truly,

L. O. Mayer

L. O. Mayer, PE
Manager, Nuclear Support Services

LOM/MHV/deb

cc: J. G. Keppler
G. Charnoff
MPCA
Attn: J. W. Ferman




SCRAM REACTIVITY FOR MONTICELLO CYCLE 5