

FROM: Niagara Mohawk Power Corporation Syracuse, New York 13202 M. H. Pratt		DATE OF DOCUMENT 12-9-69	DATE RECEIVED 12-9-69	NO.: 3815
TO: Dr Peter A. Morris		LTR. X	MEMO:	PORT:
CLASSIF: U		ORIG.: 1	CC:	OTHER:
POST OFFICE U	REG. NO:	ACTION NECESSARY <input type="checkbox"/> CONCURRENCE <input type="checkbox"/> DATE ANSWERED NO ACTION NECESSARY <input type="checkbox"/> COMMENT <input type="checkbox"/> BY:		
DESCRIPTION: (Must Be Unclassified) Ltr trans the following:		FILE CODE: 50-220 (IMPUR)		
		REFERRED TO	DATE	RECEIVED BY
		Ziemann	12-9-69	
		w/9 cys for ACTION (2 cys adv Diggs & Vollmer)		
		DISTRIBUTION:		
ENCLOSURES: CHANGE REQUEST NO. 2 to Tech Specs requesting modifications to safety limit on fuel cladding integrity.		Regulatory file		
		AEC PDR		
(3 signed & 19 conf'd cys rec'd)		Compliance (2)		
		OGC (Rm P 506 A)		
REMARKS:		H. Price & Staff		
		Skovholt		
		Dube/Levine		
		D. Thompson		
		Boyd		
		DTIE(Laughlin)		
		NSIC(Buchanan)		

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ACKNOWLEDGED

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ALL INFORMATION CONTAINED
HEREIN IS UNCLASSIFIED

DATE 10/15/2010 BY 60322 UCBAW/STP

NO FOREIGN DISSEM

NIAGARA MOHAWK POWER CORPORATION

NIAGARA  MOHAWK300 ERIE BOULEVARD WEST
SYRACUSE, N.Y. 13202

December 9, 1969



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

Dear Dr. Morris:

Re: Change Request No. 2
Provisional Operating License DPR-17
Docket 50-220

Enclosed are three signed copies and nineteen reproduced signature copies of Change Request No. 2 requesting a modification to the safety limit on fuel cladding integrity.

In accordance with Technical Specification requirement 6.1.C.2, this change has been reviewed by the Station Operations Review Committee and the Safety Review and Audit Board, respectively, and bears the approval of both bodies.

We ask that the Commission give prompt consideration to this request for change.

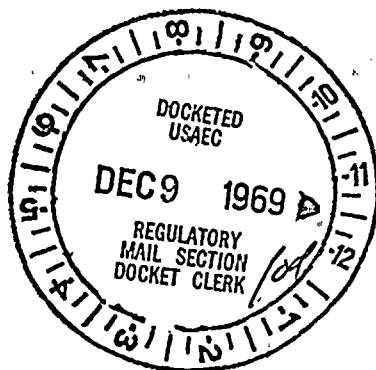
Very truly yours,



M. H. Pratt
 Vice President and
 Executive Engineer

MHP/jcl

Enclosures



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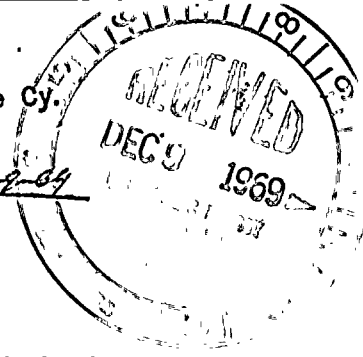
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PROVISIONAL OPERATING LICENSE DPR-17 (DOCKET 50-220)

Applicant hereby requests the Commission to change the Technical Specification (Appendix A of the above captioned license) as follows:

1. Specification To Be Changed

Safety Limit - Fuel Cladding Integrity, Figure 2.1.1 and Specification 2.1.1.b

2. Extent of Change

a. Figure 2.1.1 - Extend the 1250 psia and 1015 psia safety limit curves from 20 percent recirculation flow to 5 percent recirculation flow.

b. Specification 2.1.1.b - Modify specification to permit reactor operation in accordance with the additional data that is applicable from 5 percent to 20 percent recirculation flow as shown on the attached figure.

3. Change Requested

a. Figure 2.1.1 - Supersede the existing Technical Specification Figure 2.1.1 with the attached Figure.

b. Specification 2.1.1.b - Modify statement to read: "When the reactor pressure is less than 585 psig or reactor recirculation flow is less than 5 percent of design, the reactor thermal power shall not exceed 307 MWt."

c. References, Page 8 - Add Reference (6) Change Request No. 2 - Discussion.



4. Discussion

The extension of the fuel cladding integrity safety limit curves, Figure 2.1.1, to flows as low as 5 percent of design was performed using the same assumptions and initial conditions as were employed in the curves presently in the Technical Specifications. The design basis critical heat flux correlation given in APED-3892¹ was used. The power shape was based on a MCHFR of 1.5 at 120 percent over power and results in a total peaking factor of 3.13. Maximum feedwater temperature was assumed for the various pressures and flows and the core inlet enthalpy was based on the subcooling resulting from equilibrium operation.

Evaluation of the fuel cladding integrity safety limit in the low flow region was performed using the design basis critical heat flux correlation at the low mass flow limit. This evaluation is conservative due to the inverse relationship between mass flow and critical heat flux in the high quality region of the correlation. Critical heat flux for mass flows lower than the low limit of the design basis critical heat flux correlation are greater than that predicted by the correlation.

The proposed change of Safety Limit 2.1.1.b and Figure 2.1.1 does not reduce the margin of safety limit protection since neutron monitoring system rod block and scram trips maintain adequate margin to proposed safety limit lines on Figure

¹ E. Janssen and S. Levy, "Burnout Limit Curves for Boiling Water Reactors" APED-3892, April 1, 1962.



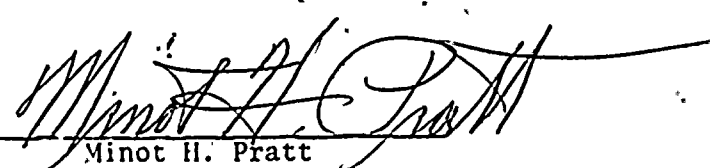
4. Discussion (Continued)

for all regions of possible operation. The core natural circulation characteristic, the lower limit of core flow for power operation, precludes operation up to the safety limit without having first reached the rod block trip and the scram trip.

Prior submittal to the FDSAR, Appendix E, has shown that the reactor is adequately protected by the pressure scram and fixed 120 percent flux scram for all transients considered such as the continuous rod withdrawal and turbine trip without bypass originating from all normal operating conditions including the low power and flow levels consistent with natural circulation.

NIAGARA MOHAWK POWER CORPORATION

by



Minot H. Pratt
Vice President
Executive Engineer

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FIGURE 2.1.1

FUEL CLADDING INTEGRITY SAFETY LIMIT

