FROM: Niagara Mohawk Power Corporation
Syracuse, New York 13202
T. J. Brosnan

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TO:
Dr. Peter A. Morris

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ACTION NECESSARY ☐
NO ACTION NECESSARY ☐
CONCURRENCE ☐
COMMENT ☐
DATE ANSWERED:
BY:

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DESCRIPTION: (Must Be Unclassified)
Ltr re our 2-11-71 ltr w/ACRS Rpt dtd 2-6-71....submitting current status on each recommendation & understandings in ACRS ltr.......

ENCLOSURES:

REMARKS:

REFERRED TO	DATE	RECEIVED BY	DATE
Ziemann W/9 cys for ACTION	3-16-71		
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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD WEST
SYRACUSE, N.Y. 13202

March 12, 1971

Regulatory File Cy.

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

Dear Dr. Morris:

Re: Docket No. 50-220

Your letter of February 11, 1971 forwarded a copy of the Advisory Committee on Reactor Safeguard's report dated February 6, 1971 pertaining to Niagara Mohawk's request for an increase from 1538 to 1850 thermal megawatts in the licensed power level of the Nine Mile Point Nuclear Station. The recommendations and understandings discussed in the ACRS letter have been carefully reviewed by our personnel and the current status of each is set forth below:

Additional Safety Valve

One additional valve identical in all respects to the original 15 valves has already been installed on the reactor vessel head.

New Reactor Scram Trips

The Technical Supplement To Petition To Increase Power Level proposes modifications to Technical Specification 2.1.2 to include the turbine stop valve closure and turbine control valve high rate of closure trips described in the ACRS letter. This specification requires both trips to be operative at all power levels above 45 percent of the 1850 thermal megawatt rating. Component installation is essentially complete; final interconnections and testing will be done before power level is raised above 1538 thermal megawatts.

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DL

Timer Settings On The Emergency Power System

Timer settings governing initiation of emergency core spray pumps will be modified to ensure a maximum core spray initiation time of 35 seconds. These modifications will be incorporated before power is raised above 1538 thermal megawatts. In addition, it is proposed that the bases for Technical Specification 3.1.4 be changed accordingly.

Refinement In Models For Evaluation Of Peak Clad Temperatures

General Electric has in preparation a topical report describing the technical bases and model assumptions used for evaluation of peak clad temperatures reached during postulated loss of coolant accidents. It is anticipated that this report will be made available to the Regulatory Staff within a few weeks. In addition, General Electric intends to continue its review of the analytical methods and possible model refinements thereto. Within approximately one year, the Regulatory Staff will be informed of the results of this review and any significant changes to previous Nine Mile Point analyses.

Reduction Of The Allowable Containment Leak Rate

The Technical Supplement To Petition To Increase Power Level proposes a modification to Technical Specification 4.3.3 incorporating this reduction.

Further Studies Of The Spent Fuel Pool

Studies of possible corrective measures to assure adequate integrity of the spent fuel pool in the postulated event of dropping a fuel cask into the pool are progressing. The results of these studies will be reported to the Regulatory Staff when completed. It is our intention that appropriate modifications be made to the Station before first use of the cask over the pool.

In-Service Inspection Of The Main Steam Lines

It is our intention to begin the expanded program for in-service inspection of the main steam lines both inside and outside the containment at the next major outage. In addition, it is proposed that Technical Specification 4.2.6 be modified to include this expanded program.

Integrity Of The Biological Shield

Analyses of the integrity of the biological shield upon postulated failure of a reactor vessel safe end have been reported by our letters of July 2, 1970 and November 23, 1970 and in the Second Addendum to the Technical Supplement To Petition To Increase Power Level. It is our understanding that these analyses demonstrate the adequacy of the as-built shield; consequently, no further action is contemplated.

Improved Leak Detection

It is our intention to install, during the next major Station outage, an atmospheric radioactivity monitoring system which will recirculate a portion of the containment atmosphere through an external loop and an air monitor.

Continued Assurance Of Reactor Pressure Vessel Integrity

Our letter of September 24, 1969, responding to an earlier ACRS report, indicated that throughout the first five years of Station operation, inspection experience will be carefully evaluated and pertinent industry-wide development efforts, such as those sponsored by the Edison Electric Institute, will be closely followed to improve means for continued assurance of pressure vessel integrity. In the course of these efforts, we intend to be particularly alert to possible improvement in access to reactor vessel surfaces for inspection purposes. It will continue to be our intent to implement to the degree practical any significantly improved inspection technique which may develop in the course of this program.

Hydrogen Control In The Containment

At the December 1970 ACRS meetings, we described active studies underway for control of the buildup of hydrogen in the containment which might follow the unlikely event of a loss of coolant accident. These studies include further review of basic data and analytical assumptions, efforts to develop a suitable containment vent procedure, and refinement of concomitant dose calculations. It is our intent to review our progress in this area with the Regulatory Staff within approximately one year.

Common Failure Modes And Failure To Scram

We are continuing our review of General Electric studies for preventing common failure modes from negating reactor scram action and consequences of failure to scram during anticipated transients. It is our understanding that a further General Electric study of a generic nature on this topic will be forthcoming within two to three months. It is our intention to review this study upon completion and to remain current on the possible ramifications of these analyses and any design features they might suggest to make tolerable the consequences of such an unlikely event for the Nine Mile Point Nuclear Station. This matter will be reviewed again with the Regulatory Staff within approximately one year.

We have previously responded to the items mentioned in the ACRS reports of April 17, 1969 and June 16, 1970 by our letters of September 24, 1969 and July 2, 1970 respectively.

Very truly yours,

T. J. Brosnan

T. J. Brosnan
Vice President and Chief Engineer