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DESCRIPTION:
Ltr furnishing addl info reg plant design modifications & suppl info reg the PSAR... Also advising that this addl info will be incorporated in an Amendment to the Appl.. w/Attached Tables 1 & 2............ & trans:

NOTE:* Received CERT OF SVC.

PLANT NAMES: Nine Mile Point, Unit 2

ENCLOSURES:
CERTIFICATE OF SERVICE showing svc of the addl info in ltr dtd 05-21-73 upon William Massar, Esq., USAEC, Washington, D. C....et...al....on 05-22-73.

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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK
300 ERIE BOULEVARD WEST
SYRACUSE, N.Y. 13202

May 21, 1973

Mr. Robert A. Clark, Chief
Gas Cooled Reactors Branch
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Clark:

Re: Nine Mile Point Unit 2
Docket No. 50-410

In accordance with recent discussions with the Commission Staff, some plant design modifications have been made and supplemental information developed in connection with the Preliminary Safety Analysis Report for our Nine Mile Point Nuclear Station Unit 2, as described below:

Table 7.2-1 - Process Pipelines Penetrating Primary Containment

Table 7.2-1 will be revised as follows: (1) A new column will be added to indicate the post-accident status, (2) Non-automatic valves which are closed and not required for post-accident function will be shown as locked closed, and (3) The "Isolation Signal" column will be further clarified.

Biological Shield Wall Design

Jet Thrust Coefficient:

In addition to the jet impingement and pressurization forces discussed in Response 5.29, the biological shield wall will be designed for the maximum loads transmitted to the wall through pipe whip restraints and the reactor pressure vessel. The time dependance of these forces will be considered. Maximum jet blowdown forces of 1.25 P·A will be used for breaks in steam lines and in water lines which are subcooled less than 22 Btu/lb.

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For water lines subcooled more than 22 Btu/lb, a maximum blowdown force of 2 P·A will be used. The effects of flow restrictors and friction losses will be considered and may be used to justify lower peak blowdown forces by reduction of jet thrust coefficients for certain breaks. Consequently, for circumferential breaks, different forces may be applied in each end of a double-ended break because of variations in flow restrictions and friction losses for the two flow paths.

Design Differential Pressure:

The maximum calculated differential pressure for the biological shield wall will be increased by a factor of 1.4 to obtain the design differential pressure. As described in Response 5.31, the peak calculated differential pressure for the preliminary design was 69 psi. A geometric loss coefficient (K) of 1.75 was used, however, the design is currently being updated and the vent area increased. Expected peak calculated differential pressure will be about 40 psi. In any event, the design differential pressure will be at least 40 percent above the peak calculated differential pressure for the final biological shield wall design.

Maximum Wetwell Temperature

The maximum wetwell temperature in the event of a large break would be 204 F, as stated in Section 14 of the PSAR.

In the event of a small break the maximum wetwell temperature is a function of the duration of the blowdown. The response to Question 5.46 shows that in the event of a small break, the wetwell pressure would reach 30 psig at about 10 minutes after the break and would reach 34 psig after 20 minutes (assuming $\frac{A}{\sqrt{K}} = 0.034$ ft.^{2}). If the wetwell pressure reached 34 psig (48.7 psia), the wetwell atmospheric temperature would be approximately 200 F. If the temperature reached 200 F, the air partial pressure would be 41.3 psia (26.6 psig) and the steam partial pressure would be about 48.7 - 41.3 = 7.4 psia.

K Factors Used in Response 5.45

The vacuum breaker system piping and vacuum breaker valves are 24 inch nominal pipe diameter. The K value for the vacuum breakers, based on manufacturer's test data for a 24 inch valve, is 1.93 (based on a 24 inch nominal pipe diameter). The total K value (6.86) for the vacuum breaker system is based on 24 inch piping. Therefore, when using the K value of 6.86, a

pipe diameter of 24 inches should be used, for which the flow area is 3.14 ft^2.

The throat diameter of the vacuum breaker is 19.4 inches and has an area of 2.05 ft^2. If this area is used as the flow area, the K value must be adjusted. Crane[(1)] shows that

$$K_2 = K_1\left(\frac{d_2}{d_1}\right)^4 .$$

Thus, in this case, K would be: $K=6.86 \left(\frac{19.4}{24}\right)^4=2.93$, and the required number of vacuum breakers would remain the same.

Codes Referenced in PSAR Section 12

All codes in Section 12 will be reviewed and updated as necessary.

Reactor Trip Control Coil

The fifth sentence in the response to part (b) of Request 7.2 will be revised to read as follows:

"The manual initiation logic output contacts will be introduced into the reactor trip contactor coil circuits."

Design of Building Exterior Walls and Foundations

Elevation 263 ft. is the height of the dike. The design water level for structures is based on the grade El. 260 ft. plus a water accumulation of 0.35 ft. due to the maximum probable rainfall. All seismic Class I building floor slabs will be at El. 261 ft. or higher. The maximum probable flood level of Lake Ontario at the site is well below this level. Exterior walls and foundations will be designed for a hydrostatic head to El. 261 ft. Buoyant forces are considered for all structural design.

Protection of Engineering Safety Systems

Postulated failure of non-seismic Class I equipment will not flood or impair any engineering safety system function.

---

(1) Crane Technical Paper 410, Flow of Fluids Through Valves, Fittings and Pipe, 1965.

Unprocessed Waste Collector Tank Analysis

Request 9.6, page 9.6-2, Item C will be revised to read as follows:

"C. Safety Guide 3 meteorology is used; i.e., ground level release for the reactor building and for the radwaste building."

Seismic Design of Structures

PSAR Section 12.3-9 will be revised to include the following:

The seismic loading referred to in paragraph 1.5.6 of the American Institute of Steel Construction Specification will be considered as the Operating Basis Earthquake. The 1/3 increase in stresses allowed by the American Institute of Steel Construction specifications for loading conditions that include the Operating Basis Earthquake will not be taken.

The 25 percent reduction in load factors allowed by the American Concrete Institute 318-71 for Operating Basis Earthquake loading conditions will not be taken. The following load factor equations for the Operating Basis Earthquake will be used:

$$U = 1.4D + 1.7L + 1.9\ OBE$$

$$U = 0.9D + 1.9\ OBE$$

where

D = dead load

L = live load

OBE = Operating Basis Earthquake load

Seismic Design of Equipment and Structures

The PSAR will be expanded to include loading criteria for all Category I equipment and buildings in addition to the primary containment.

Load Combination for Pipe Break Analyses

Load combination equations in the PSAR will be revised to include the following loads from postulated pipe breaks: (1) thermal loads, (2) anchor loads, (3) pipe whip loads, (4) pressure loads, and (5) jet impingement loads.

Missile Analyses

The penetrations for the one-inch diameter rod provided in Response 12.6 were based on the U. S. Army Ballistics Research Laboratory formula. This formula gives no basis for determining the thickness of concrete required to prevent spalling. The penetrations for the one-inch diameter rod provided in Response 12.19 are based on the Armann and Whitney and the National Defense Research Committee formulae. These formulae were used for the tornado generated missiles since they account for concrete wall thickness required to prevent spalling. The two methods give somewhat different results when considering depth of penetration for similar missile velocities. Responses 12.6 and 12.19 provide formula references. The missiles are generated from two sources: tornadoes or internally generated. Thus, their velocities are different at impact on the concrete wall.

Composite Damping in a Coupled System

The response to Request C.3.4 will be revised to add the following statement at the end of the response:

"When the damping of the system cannot be reasonably approximated by this method, more exact methods are used."

Control Room Ventilation

The response to Request 10.42 will be revised to specify that redundant smoke detectors and radiation monitors in the outside air intake duct will be used to automatically switch the control room air conditioning system to the emergency mode should smoke or high radiation be detected in the intake ducts.

Maximum Probable Precipitation

The response to Request 2.33 will be revised to indicate that while roof and site drainage will be designed for a local maximum precipitation rate of 6.0 inches in one hour, analyses will be performed to verify that a precipitation rate of 8.4 inches in one hour will not affect any safety related function.

Application of Inelastic Analysis

With reference to Applicant's previous response to Request C.7.11, General Electric will obtain AEC approval and

concurrence prior to application of inelastic system analyses. This commitment is in accord with stipulations relative to Faulted Conditions and the use of ASME Appendix F contained in Dr. J. M. Hendrie's July 12, 1972 letter to Mr. A. P. Bray.

Valve Operability

The response to Request 10.57 will be revised to add the following:

A table will be provided in the FSAR which will identify all active valves with a safety function. The following information will be provided for each valve: (a) Valve identification, (b) Valve operation (manual or automatic), (c) Valve design conditions (codes and environmental), (d) Valve operation during a safety function (i.e., does it open or close), (e) Test condition that the valve or a similar valve was subjected to, if applicable. If the valve was qualified by conservative analysis in lieu of testing, this will be noted and the analysis discussed, and (f) Remarks (i.e., statement as to what assurance there is that the valve will function; for example, past experience, etc.).

Analysis of Reactor Building Wall and Mat Intersection

Section 12.5.3 (page 12.5-5) of the PSAR will be revised to include the following:

The concrete wall close to the mat will be conservatively assumed to be cracked horizontally to the neutral axis of the transformed section for any subsequent loading condition. For vertical cracks the following two assumptions will be utilized for the analysis and the most conservative results will be used for the design:

(1) Completely cracked vertically
(2) Vertically uncracked for the lower five feet of wall

Battery Charger Capacity

Part (b) of the response to Request 8.16 will be revised to indicate that the battery charger will be able to supply all DC loads and recharge the battery from the fully discharged condition within 24 hours.

Standby Diesel System Testability

The response to Request 8.17 will be revised to include the following:

"While the detailed testing circuitry has not yet been designed, it will provide testability of the onsite emergency power sources (Diesel Generators) at power. This testability will include the starting and load sequencing logic required under emergency conditions."

Standby Gas Treatment System

PSAR Section 5.3.4 and Figure 5.3-2 will be revised to include a high efficiency particulate air prefilter in each train of the standby gas treatment system.

System Level Bypass Indication

Parts a and b of Response R7.18 will be modified as follows:

"Automatic indication will be provided in the control room to inform the operator that a system is inoperable. Annunciation will be provided to indicate a system or part of a system is not operable. For example, the reactor protection (trip) system and the containment and reactor vessel isolation system have annunciators, lighting and sounding whenever one or more channels of an input variable are bypassed. Bypassing is not permitted in the trip logic or actuator logic. The HPCS, LPCS, RHRS, and the ADS have keylock operating bypasses and test bypasses actuated by the insertion of test jacks. Any operating or test bypass will annunciate at the system level: e.g., HPCS "in test". Bypass of certain infrequently used pieces of equipment, such as manual locked-open valves, are not automatically annunciated in the control room. However, capability for manual actuation of each system level bypass indicator is provided in the control room for those systems that do have infrequently used bypasses. A keylock switch is used for this manual actuation. Examples of automatic indication of inoperability are given in the Response to Request 7.9."

[illegible]

Quality Assurance (QA)

QA Manager:

The QA Manager reports to the Vice President-Engineering on matters involving design and construction. He reports to the Vice President-Electric Operations on matters involving operations. The point of trade-off between the Vice President-Engineering and the Vice President-Electric Operations occurs at the point in time at which the plant goes into commercial operation. As a practical matter, however, both Vice Presidents are kept advised of activities involving the Unit during both phases.

QA Program:

The QA program as described in the application is in effect for work now in progress involving safety-related equipment. Frequent design review meetings are being held with representatives of the organizations involved in producing the Unit design. These meetings ensure that applicable regulatory requirements, codes, standards and design bases are correctly translated into specifications, drawings, procedures and instructions. Included also in deliberations during the design review meetings are resolutions regarding QA/QC adequacy, design interfaces and Regulatory Guides.

The QA activities which have been initiated include assessments of the adequacy of the programs of the NSSS and nuclear fuel supplier, the architect-engineer, and the reactor pressure-vessel supplier. Quality assurance audits have already been conducted on the NSSS and nuclear fuel supplier, and on the architect-engineer with particular emphasis on the design activities currently being undertaken. These activities include such items as design control, procurement document control, the preparation of instructions, procedures and drawings, document control, control of purchased material, equipment and services, QA records, audits, etc.

Staff Qualifications:

Attached Tables 1 and 2 illustrate, respectively, the minimum qualifications of QA personnel as established by the NMPC QA manual, and the actual qualifications of personnel currently assigned.

QA Involvement:

Niagara Mohawk Power Corporation QA organization's primary involvement is to assure adequate information on quality and safety requirements by audit of the design and procurement functions of NMPC project management, the architect-engineer constructor and the NSSS and fuel supplier (General Electric).

GE-NED incorporates a QA organization to ensure that the requirements of 10 CFR 50, Appendix B, are met for all of its nuclear projects, including Nine Mile Point Nuclear Station - Unit 2. In order to establish bases for audit of GE-NED activities, the NMPC QA organization is reviewing safety-related component procedures and documentation. The NMPC QA organization is also conducting audits of the GE-NED activities both to assess the adequacy of its program relative to 10 CFR 50, Appendix B, and to verify its compliance thereto.

S&W, the architect-engineer, also maintains a QA organization to ensure that the requirements of Appendix B are met. In order to establish bases for audit of S&W activities, the NMPC QA organization is reviewing procedures and documentation, and is attending internal audits of S&W activities. The NMPC QA organization is also conducting audits of S&W activities both to assess the adequacy of its program relative to 10 CFR 50, Appendix B, and the NMPC QA Manual, and to verify its compliance thereto.

As part of these reviews and audits, the NMPC QA organization makes inquiries pertinent to compliance with the safety requirements imposed by the Code of Federal Regulations and the PSAR.

NMPC conducts design review meetings on safety-related systems and components. A prerequisite to the consideration of a particular system or component at these meetings is that all necessary prior reviews, approvals, etc. be evident in the documentation. Interface meetings are in accordance with procedures which include notification of required attendees.

Tests and inspection notification to S&W and GE-NED are being required of vendors by the purchase documentation. The procedures for subsequent notification of NMPC by S&W and GE-NED will be established in the near future.

NMPC QA is notified routinely of internal engineering audits performed by S&W. In order to establish bases for audits

and for familiarization with S&W activities, NMPC has attended these audits. Audits of vendors' shops will be initiated on Unit 2 components.

The information described above will be incorporated in an Amendment to our application and submitted to the Commission in the near future.

Very truly yours,

Philip D Raymond

Philip D. Raymond
Vice President-Engineering

GKR/vk

TABLE 1

MINIMUM STAFF QUALIFICATIONS

Title	*Min Required Degree	Min Years Experience	Required Background
Manager of Quality Assurance	BS	10	At least ten years in charge of responsible assignments including several functions in the design, construction or operation of a nuclear facility.
Project QA Supervisor	BS	10	At least ten years of quality related work or equivalent experience in the design, construction or operation of a nuclear facility.
Senior QA Engineer	BS	5	At least five years of quality related work or equivalent experience in the design, construction or operations of a major industrial facility.
QA Engineers	BS	2	At least two years experience in quality related functions in the design, construction, or operation of a major industrial facility.
Technical & Support Personnel	High School	0	Education in a technical discipline.

*Or equivalent qualifications

TABLE 2

ACTUAL STAFF QUALIFICATIONS

Title	Education	Registration	Experience
QA Manager	BSEE	PE (NYS)	24 yrs. (Total); Engr. (Nuclear) - 10 yrs.
QA Supervisor	BSEE	PE (NYS)	Operation - 5 yrs.; Engr. - 20 yrs. (Total); Engr. (Nuc.) - 10 yrs.
QA Supervisor	BSME,MSNE	PE (NYS): PE (MICH)	Engr. - 20 yrs. (Total); Engr. (Nuclear) - 17 yrs.
QA Engineers -			
A	BSEE	-	Transm. & Distr. Operations- 14 yrs., of which 50% involved related engineering
B	BSChE	PE (NYS)	Engr. 22 yrs. (Total); Engr. (Nuclear) - 1 yr.
C	BSCE	CE (OH) SE (IL) PE (NYS)	Engr. (Constr.) - 18 yrs. (Total)
D	University of Buffalo. 5 yrs. parttime	PE (NYS) PE (PA)	Design & Engr. - 35 yrs. (Total); Engr. (Nuclear) - 10 yrs.
E	BSEE, MSEE	-	Engr. 3 yrs. (Total)
F	BS (Metallurgy); MSME	-	Engr. 19 yrs. (Total); Engr. (Nuclear) 19 yrs.
G	BS (Metallurgy)	-	Engr. 21 yrs. (Total)

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BEFORE THE UNITED STATES

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
Niagara Mohawk Power Corporation	)	Docket No. 50-410
(Nine Mile Point Unit 2)	)	

Regulatory File Cy.
Received w/Ltr Dated 05-21-73

CERTIFICATE OF SERVICE

I hereby certify that I have served, pursuant to the Atomic Energy Commission's Rules and Regulations, copies of the Applicant's letter of May 21, 1973 concerning the process pipelines penetrating primary containment and other matters in the above-captioned proceeding upon the following persons this 22nd day of May 1973.

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