MAR 18 1986

MEMORANDUM FOR: Elinor G. Adensam, Director, BWR Project Directorate #3
Division of BWR Licensing

FROM: F. Rosa, Chief
Electrical, Instrumentation & Control Systems Branch
Division of PWR Licensing-A

SUBJECT: INSTALLATION OF A CATEGORY 1 (IN ACCORDANCE WITH R.G. 1.97) NEUTRON FLUX INSTRUMENTATION AT NINE MILE POINT UNIT 2

As requested by the Nine Mile Point Unit 2 Project Manager (M. Haughey) the purpose of this memorandum is to document our position on the installation of the subject instrumentation at NMP-2. Because this may be an appeal item Ms. Haughey requested this memorandum ahead of our formal SER so that she can pursue this item with the applicant in a timely manner prior to licensing.

R.G. 1.97 identifies neutron flux as one of the variables required to be monitored during and following an accident. The design and qualification criteria for the associated instrumentation is listed as Category 1 in the guide. One of the qualification criteria for a Category 1 instrument as defined by R.G. 1.97, is that it be environmentally qualified. The applicant indicated in his January 20, 1986 letter that the environmental qualification of his neutron flux instrumentation under harsh environment conditions is for a limited time. In subsequent conversations with him he indicated that the amount of time the instrumentation could survive in a harsh environment is only a matter of seconds. It is our position that, in accordance with R.G. 1.97, the environmental qualification of this instrumentation should ensure its survivability during and following an accident.

Because a Category 1 neutron flux instrument is presently not available but is an industry development item, we have been requiring utilities to follow the development of this hardware and install a Category 1 instrument when it becomes available. We have presently not received a commitment from the NMP-2 applicant to install a Category 1 instrument when it becomes available. We therefore intend to condition the NMP-2 license that this be done. Our position has been discussed with the NMP-2 applicant.

Original Signed By
Faust Rosa*

Faust Rosa, Chief
Electrical, Instrumentation &
Control Systems Branch
Division of PWR Licensing-A

cc: See attached list

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cc: C. E. Rossi
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LP 25

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

March 3, 1986
(NMP2L 0644)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

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

Attached is additional information concerning the Nine Mile Point Unit 2 compliance to Regulatory Guide 1.97. This response is in addition to the response provided on January 20, 1986.

This material was discussed in telephone conversations with your staff on February 13, 1986 and February 21, 1986. Niagara Mohawk expects that this material provides the necessary information to close Confirmatory Item 10.

Very truly yours,

C. V. Mangan
Senior Vice President

TL:ja
1361G
Attachment

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

8603070317
1P

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)

Niagara Mohawk Power Corporation) Docket No. 50-410

(Nine Mile Point Unit 2))

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Senior Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

C V Mangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 3rd day of March, 1986.

Notary Public in and for
Onondaga County, New York

My Commission expires:

Notary P...
Qualified in ...
My Comm... 87

2. Neutron flux - The applicant should provide Class 1E power sources for this instrumentation; the applicant should show that the source and intermediate ranges have sufficient overlap.

Response

In addition to the information provided in our January 20, 1986 letter, Niagara Mohawk will continue to follow and evaluate developments in the nuclear industry concerning neutron flux monitoring instrumentation. Prior to the conclusion of the first refueling outage, Niagara Mohawk will update and identify to the Nuclear Regulatory Commission the status of the neutron flux monitoring instrumentation issue for Nine Mile Point Unit 2.

4. Coolant level in reactor - The applicant should identify the remainder of this instrumentation in accordance with Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations identified.

Response

As stated in our January 20, 1986 letter, conformance to Regulatory Guide 1.97, Revision 3 is accomplished by the use of two transmitters per division, one fuel zone and one wide range. As noted in Section 3.3.3 of your November 15, 1985 letter, the wide range transmitters were omitted from Table 421.36-1 and will be added in the next Final Safety Analysis Report amendment.

The wide range transmitters (2ISC*LT9C and D) are calibrated to monitor the 375.70 in. level, which is inside the fuel zone transmitter range, to the 585.70 in. level which is 62.3 in. below the centerline of the main steam lines at 648 in. It should be noted that all safety trips from reactor level occur within these level ranges.

This range meets the intent of the regulatory guide which is to restore and maintain reactor pressure vessel water level to ensure adequate core cooling.

Water level indication is available in the control room to the operator from one transmitter (2ISC*LT105) covering the 525.70 to 925.70 in. level which is well above the main steam lines. A second transmitter (2ISC*PDT110) covering the 525.70 to 705.70 in. level is also available to the control room operator on a strip chart recorder. However, neither of these transmitters (2ISC*PDT110 and LT105) fully meet the qualification requirements of Regulatory Guide 1.97.

LP-25

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

January 20, 1986
(NMP2L 0589)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

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

Attached is the Nine Mile Point Unit 2 response to the letter from W. Butler (NRC) to B. G. Hooten (NMPC), dated November 15, 1986 concerning conformance to Regulatory Guide 1.97.

This material provides the information necessary to close Confirmatory Item 10.

Very truly yours,

C. V. Mangan
Senior Vice President

TRL:ja
Attachment

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

8001240044
8 pp.

ATTACHMENT

1. The licensee should identify plant-specific Type A variables and verify that the instrumentation for them is Category I.

Response

Table 421.36-1 of the Nine Mile Point Unit 2 Final Safety Analysis Report will be revised to identify the following Type A variables.

1. Containment hydrogen concentration (2) (same as variables C11a and b)
2. Containment oxygen concentration (2) (same as variables C12a and b)
3. Reactor vessel pressure (2) (same as variables B6a and b)
4. Reactor vessel level (4) (same as variables B4a, b, c and d)
5. Suppression pool water temperature (8) (same as variables D6a and b)
6. Drywell atmosphere temperature (18) (same as variables D7a and b)
7. Drywell atmosphere pressure (2) (same as variables D4a and b)

2. Neutron flux - The applicant should provide redundant Class 1E power sources for this instrumentation; the applicant should show that the source and intermediate ranges have sufficient overlap.

Response

The source of power (see Nine Mile Point Unit 2 Final Safety Analysis Report Figure 8.3-10) for Source Range Monitors (SRMs) and Intermediate Range Monitors (IRMs) originates from reliable normal dc sources. Normal supply originates from stub buses (2NJS-US6 and US5) to 24/48-V dc distribution panels (2BWS-PNL300A and 2BWS-PNL300B). A normal power source feeds two battery chargers per division that service the 24/48-V dc distribution panel(s) and maintain the charge on two 24-V dc batteries. The batteries are available to service the 24/48-V dc distribution panels on loss of normal power.

The power source for Low Power Range Monitor (LPRM) groups and Average Power Range Monitor (APRM) channels is from the RPS/UPS channelized Divisions 1 through 4. This power is fed to RPS buses by means of a UPS which has normal, alternate and battery backup sources. Power sources are channelized RPS Divisions 1 through 4. The power distribution system for this instrumentation is described in detail in Section 8.3.1.1.3 of the Final Safety Analysis Report.

It is Niagara Mohawk's determination that this design provides reliable power sources.

Although the power supplies are classified as nonsafety-related, this does not impair the ability of the neutron monitoring instrumentation to perform its required detection and trip functions. The trip function is configured to trip and initiate a scram on loss of power. The instrumentation is seismically and environmentally qualified, so the trip function is ensured on loss of power.

The operating ranges of the source range (SRM) and intermediate range (IRM) devices are as follows:

SRM = 1×10^3 to 1.5×10^9 nv
IRM = 1×10^8 to 1.5×10^{13} nv

The overlap of the ranges is 1×10^8 to 1.5×10^9 nv.

Additionally, there is a typographical error in Table 421.36-1. The lower end of the IRM range should be 4.0×10^{-5} percent power. This will be corrected in the next Final Safety Analysis Report amendment.

3. Reactor coolant system soluble boron concentration - The applicant should identify the range of the instrumentation being supplied for this variable.

Response

The range is 50 to 2,000 ppm boron in solution. This information will be incorporated in the next Final Safety Analysis Report update.

4. Coolant level in reactor - The applicant should identify the remainder of this instrumentation in accordance with Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations identified.

Response

Conformance to Regulatory Guide 1.97, Revision 3, is accomplished by the use of two transmitters per division, one fuel zone and one wide range. As noted in Section 3.3.3 of your November 15, 1985 letter, the wide range transmitters were omitted from the table and will be added in the next Final Safety Analysis Report amendment.

The wide range transmitters (2ISC*LT9C and D) are calibrated to monitor the 375.70 in. level, which is inside the fuel zone transmitter range, to the 585.70 in. level which is 62.3 in. below the centerline of the main steam lines at 648 in.

This range is considered to meet the intent of the regulatory guide which is to restore and maintain reactor pressure vessel water level to ensure adequate core cooling.

5. Drywell pressure - The applicant should provide independent Class 1E power sources for these instrument channels.

Response

The two instrument channels are powered from separate Class 1E sources. Table 421.36-1 is incorrect and will be corrected in a future Final Safety Analysis Report amendment.

6. Drywell sump level - The applicant should provide instrumentation for this variable.

Response

See Item 7.

7. Drywell drain sumps level - The applicant should provide instrumentation for this variable.

Response (Items 6 and 7)

The drywell sump level instruments provide indication of identified and unidentified leakage during normal operating conditions. The instrumentation, which is nonsafety grade (Category 3), is located on drain tanks in the secondary containment, outside of the drywell. Under accident conditions, these drain tanks are automatically isolated from the primary containment to prevent the escape of any post-accident reactor fluid from the drywell. In this situation, the drywell sump level indication is no longer meaningful and thus serves no post-accident safety function.

Other instrumentation is available to identify leakage into the drywell. This includes drywell pressure, drywell temperature, and containment radiation. These instruments meet the Category I requirements of Regulatory Guide 1.97.

8. Radiation level in circulating primary coolant - The applicant should supply the recommended instrumentation and the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations from the regulatory guide, and justify those deviations.

Response

This instrumentation is not provided at Nine Mile Point Unit 2.

Justification

The usefulness of information obtained by monitoring the radiation level in the circulating primary coolant, in terms of helping the operator in his efforts to prevent and mitigate accidents has not been substantiated. The particular planned operator action to be taken based on monitoring this variable is not specified in the current draft of the Emergency Procedures. The critical actions taken to prevent and mitigate a gross breach of fuel cladding are to shutdown the reactor and maintain water level. Monitoring primary coolant radioactivity has no influence on either of these actions. The purpose of this monitor falls in the cateogry of "information that the barriers to release of radioactive material are being challenged" and

"identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." Additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables, have not been identified.

Regulatory Guide 1.97 specifies measurement of the radioactivity of the circulating primary coolant as the key variable in monitoring fuel cladding status during isolation of the nuclear steam supply system (NSSS). The words "circulating primary coolant" are interpreted to mean coolant, or a representative sample of such coolant, that flows past the core. A basic criterion for a valid measurement of the specified variable is that the coolant being monitored is coolant that is in active contact with the fuel, i.e., flowing past the failed fuel. Monitoring the active coolant (or a sample thereof) is the dominant consideration. The post-accident sampling system (PASS) provides a representative sample which can be monitored.

The concern of Regulatory Guide 1.97 assumes a situation in which the NSSS is isolated and the reactor is shutdown. This assumption is justified because the monitors in the off-gas system and main steam tunnel provide reliable and accurate information on the status of fuel cladding when the plant is not isolated. Further, the PASS, once activated, provides an accurate status of coolant radioactivity and hence, cladding status. In the interim between NSSS isolation and operation of the PASS, monitoring of the primary containment radiation and hydrogen levels provides information on the status of the fuel cladding.

Present emergency procedures provide that once initial core damage is estimated using information obtained from the analysis of PASS samples, the estimate is confirmed using containment hydrogen analysis, containment high-range radiation monitoring, water level indications, and Sr, Ba, La, and Ru analyses. Therefore, no Type C Category I instrumentation is provided to measure the subject variable.

The Niagara Mohawk position agrees with the BWR Owners Group position on this variable.

9. Analysis of primary coolant - The applicant should identify the range of the instrumentation being supplied for this variable.

Response

The Instrument range is 10^{-6} to 10^{1} Ci/gm and will be incorporated in the next Final Safety Analysis Report update.

10. Radiation exposure rate - The applicant should show that the ranges encompass the expected radiation levels in their locations.

Response

Access is not required in any area of secondary containment to service safety-related equipment in a post-accident situation. When accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere.

Area monitors provided in areas outside secondary containment where access may be required post-accident have ranges that envelop the dose rates expected in these areas at the time access is required.

11. Residual heat removal heat exchanger outlet temperature - Environmental qualification should be addressed in accordance with 10CFR50.49.

Response

The instrumentation for this variable is Category I and environmentally qualified to 10CFR50.49. It is in compliance with the requirement for Regulatory Guide 1.97, Category 2 instrumentation. Table 421.36-1 of the Final Safety Analysis Report is incorrect and will be corrected in a future amendment.

12. Cooling water temperature to engineered safety feature system components - The applicant should justify the deviation in range.

Response

The temperature range of 2SWP*TE31A and B complies with the intent of Nuclear Regulatory Commission Regulatory Guide 1.97, Revision 3. The intent of the regulatory guide is to ensure that instrument ranges are selected so that the instrument will always be on scale. 2SWP*TE31A and B are located on the service water supply header and are used to monitor service water supply temperature. The temperature range of service water instrument is based upon the range for Lake Ontario, which normally varies from 38°F to 77°F and which is well within the instrument range of 32°F to 130°F.

13. Secondary containment area radiation - The applicant should supply the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations; environmental qualification should be addressed in accordance with 10CFR50.49.

Response

Environmentally qualified area monitors with ranges of 10^{-1} to 10^{4} R/hr are not provided in secondary containment at Nine Mile Point Unit 2.

Justification

The use of local area radiation monitors to detect breach or leakage through primary containment penetrations is inappropriate. In general, radiation levels in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in emergency core cooling system (ECCS) piping. Localized hot spots due to piping sources and primary containment penetrations and hatches will provide ambiguous indications. Breach of primary containment will be detected by the reactor building exhaust gaseous effluent monitor prior to the reactor building isolation and the noble gas channel of the main stack gaseous effluent monitor following the isolation of the reactor building. Therefore, these area monitors are not necessary and are not implemented at Nine Mile Point Unit 2.

In the long term, accessibility to the secondary containment will be reestablished using a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere.

14. Noble gas, radwaste vent - The applicant should either provide the recommended range or justify the use of the lesser range.

Response

See Item 15.

15. Noble gas, common plant vent - The applicant should either provide the recommended range or justify the use of the lesser range.

Response (Items 14 and 15)

The noble gas channels of the gaseous effluent monitors for the radwaste/reactor building vent and the main stack release points have design ranges of 10^{-6} to 10^{4} uCi/cc, which meet Regulatory Guide 1.97 requirements.

16. Plant and environs radiation - The applicant should identify the ranges of this instrumentation and show that the ranges are adequate.

Response

Two types of portable radiation detection and instrumentation are provided to monitor the plant and environs. An ion chamber detector with a range of 10^{-3} to 50 R/hr is used for low-level gamma and beta radiation monitoring. A Geiger-Muller Teletector type detector with a range of 10^{-2} to 10^{3} R/hr is used for high level gamma radiation monitoring. With a combined range of 10^{-3} to 10^{3} R/hr, these instruments have adequate range to envelop the dose rates expected outside the plant buildings after an accident.

17. Accident sampling (primary coolant, containment air and sump) - The applicant should provide the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations from the regulatory guide, and justify those deviations.

Response

1. Instrument range - Analysis range is given in Table II.B.3-2 (See Section 1.10, Table II.B.3-2 of Nine Mile Point Unit 2 Final Safety Analysis Report). The ranges meet or exceed requirements of Regulatory Guide 1.97, within instrument limitations, with the exception of the dissolved gas sample analysis. The ranges given for dissolved gas analysis were approved by the Nuclear Regulatory Commission in a letter to General Electric (letter from W. Johnston [Nuclear Regulatory Commission] to G. Sherwood [General Electric] dated July 17, 1984).

2, 3, 4, 6. Environmental qualification, seismic qualification, quality assurance, and power supply - These have been addressed in Table 421.36-1 and meet the requirements of Regulatory Guide 1.97.

5. Redundancy and sensor locations - This is not applicable, since analysis is done in a chemistry laboratory on grab samples. Regulatory Guide 1.97 has no specific provision.

7. Display location - This is not applicable, since analysis is done in a chemistry laboratory, the display is on each individual instrument. This meets the requirements of Regulatory Guide 1.97.

18. Primary containment isolation valve position - The applicant should provide justification for the exemption of instrumentation for the traversing incore probe system isolation valves.

Response

The traversing incore probe (TIP) system isolation valves consist of ball valves, operated when the probe is out of the guide tube, and shear valves manually operated if the probe is in the guide tube.

The TIPs are normally withdrawn and the ball valves are closed. If an event occurs while the TIP is inserted into the core and the TIP should fail to retract, the shear valve can be operated manually to provide the necessary containment isolation.

These valves are classified nonessential and are provided with non-Class 1E automatic isolation signals and power and as such, cannot meet Regulatory Guide 1.97, Category 1 or 2 requirements. (For further explanation, refer to the Nine Mile Point Unit 2 Final Safety Analysis Report Section 1.10, TMI action item II.E.4.2 concerning the Containment Isolation Dependability.)

Additionally, any leakage through this line has been incorporated in the radiological LOCA analysis of Chapter 15.6.5 of the Nine Mile Point Unit 2 Final Safety Analysis Report.

19. Please verify that Category I instrumentation is or will be provided for neutron flux instrumentation (from W. R. Butler letter of October 15, 1985). (Table 421.36-1 of Final Safety Analysis Report commits to environmentally and seismically qualified equipment.)

Response

Neutron flux measurement devices located in harsh environment areas are environmentally and seismically qualified for the anticipated environments. The environmental qualification under harsh environment conditions is for a limited time, but the time is sufficient to perform the detection, mitigation, and monitoring functions required of the instrumentation. Instrumentation located in mild environment areas is seismically qualified.

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EISENHUT,D.G. Division of Licensing

SUBJECT: Amend 14 to OI application,consisting of Amend 14 to FSAR.
Affidavit encl.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

OCTOBER 5, 1984
(NMP2L 0169)

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

Re: Amendment 14 to Application to Operating License
Nine Mile Point Unit 2
Docket No. 50-410

In accordance with 10CFR50.30 (c) (1), 10CFR51.24 and your March 29, 1983 letter, enclosed are three originals and 60 copies of Amendment 14 to the Final Safety Analysis Report. These changes incorporate certain responses into the text of the Final Safety Analysis Report as appropriate. Also included are changes which have resulted from our continuing review of these documents.

Very truly yours,

C. V. Mangan
Vice President
Nuclear Engineering & Licensing

JM:ja
Enclosure
xc: Project File (2)

B001

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

C. V. Mangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 24th day of September, 1984.

Christine Austin
Notary Public in and for
Onondaga County, New York

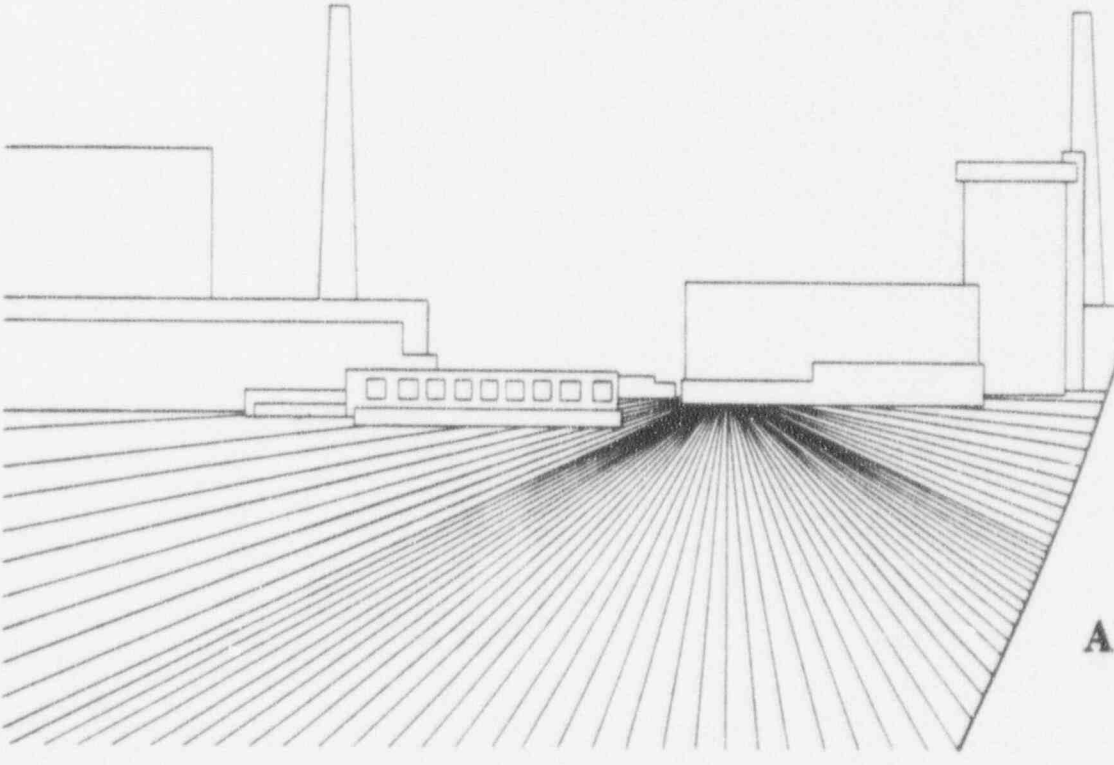
My Commission expires:

CHRISTINE AUSTIN
Notary Public in the State of New York
Qualified in Onondaga Co. No. 4787687
My Commission Expires March 30, 19[illegible]

FINAL SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION – UNIT 2

NIAGARA MOHAWK



AMENDMENT 14

20PP

QUESTION F421.36 (7.5)

The NRC staff has recently issued Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" via Supplement 1 to NUREG-0737. This Reg. Guide revision reflects a number of major changes in post-accident instrumentation. Supplement 1 to NUREG-0737 includes specific Reg. Guide 1.97 implementation requirements for plants in the operating license review stage.

Provide a description of how the Nine Mile Point Unit 2 design conforms to the provisions of Reg. Guide 1.97, Revision 2. This description should be in the form of a table that includes the following information for each Type A, B, C, D, E variable shown in Regulatory Guide 1.97:

(1) instrument range

(2) environmental qualification (as stipulated in guide or state criteria)

(3) seismic qualification (as stipulated in guide or state criteria)

(4) quality assurance (as stipulated in guide or state criteria)

(5) redundancy and sensor(s) location(s)

(6) power supply (e.g., Class 1E, non-Class 1E, battery backed)

(7) location of display (e.g., control room board, SPDS, chemical laboratory

Deviations from the guidance in Reg. Guide 1.97 should be explicitly shown, and supporting justification or alternatives should be presented.

RESPONSE

See Table 421.36-1. 14

TABLE 4Q21.36-1

CONFORMANCE TO REGULATORY GUIDE 1.97

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
B13-D193	Power Rng Flux Level	B1a.	1	Core	0.5-125% pwr	Yes	Yes	II	Non-1E	P603	1
	Average Pwr Rng Flux Lvl	B1b.	1	N/A	0-125% pwr	No	No	II	Non-1E	P603	2
C51-N002A-H	Inter-mediate Rng Flux Level	B1c.	1	Core	40x10-6 - 12.6% Pwr	Yes	Yes	II	Non-1E	P603	-
C51-N001A-D	Source Rng Flux Level	B1d.	1	Core cps	0.1-1x10^6	Yes	Yes	II	Non-1E	P603	-
	Control Rod Position	B2	3	Core	Withdrawn or Scram	Yes	Yes	II	Non-1E	P603	-
	Rx Coolant Boron Conc	B3	3	Unit 1 H.P. Lab.	(ltr)	(ltr)	(ltr)	II	Non-1E	-	4
2ISC*LT13A/ B22-N044A	Reactor Vsl Level - A (Fuel Zone)	B4a.	1	Rx Bldg (Sec Contmt)	230.64-430.69"	Yes	Yes	I	Div. 1	P601	5
2ISC*LT13B/ B22-N044B	Reactor Vsl Level - B (Fuel Zone)	B4b.	1	Rx Bldg (Sec Contmt)	230.69-430.69"	Yes	Yes	I	Div. 2	P601	5
N/A N/A	Core Temperature	B5	1	-	-	-	-	-	-	-	6
2ISC*PT6A/ B22-N062A	Reactor Vsl Pressure-A	B6a.	1	Rx Bldg (Sec Contmt)	0-1500 psig	Yes	Yes	I	Div. 1	P601	-
2ISC*PT6B/ B22-N062B	Peactor Vsl Pressure-B	B6b.	1	Rx Bldg (Sec Contmt)	0-1500 psig	Yes	Yes	I	Div. 2	P601	-

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TABLE Q21.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CMS*PT2A	Drywell Pressure-A	B7a.	1	Rx Bldg (Sec Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P601	-
2CMS*PT2B	Drywell Pressure-B	B7b.	1	Rx Bldg (Sec Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P898	-
2CMS*PT7A	Suppression Chamber Pressure-A	B7c.	1	Rx Bldg (Sec (Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P601	-
2CMS*PT7B	Suppression Chamber Pressure-B	B7d.	1	Rx Bldg (Sec Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P898	-
See Note 7	Drywell Sump Level	B8	1	-	-	-	-	-	-	-	7
2CMS*PT1A	Primary Containment Pressure-A	B9a.	1	Rx Bldg (Sec Contmt)	-5 to +5 psig	Yes	Yes	I	Div. 1	P601	8
2CMS*PT1B	Primary Containment Pressure-B	B9b.	1	Rx Bldg (Sec Contmt)	-5 to +5 psig	Yes	Yes	I	Div. 2	P601/ P898	8
2AAS*MCV134, 135	Primary Containment Vlv Isolation - AAS	B10a1	1	N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2AAS*MCV136, 137	Primary Containment Vlv Isolation - ASS	B10a1	1	N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2CCP*MOV17A, B;124, 265	Primary Containment Vlv Isolation - CCP	B10b1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602/ P873	37

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CCP*MOV16A, B; 90A, B; 122, 273	Primary Containment Vlv Isolation - CCP	B10b2	1	N/A	N/A	Yes	Yes	I	Div. 2	P602/ P873	37
2CMS*SOV24A, C;26A,C;32A; 33A;34A;35A; 60A,B;63A,B	Primary Containment Isolation - CMS	B10c1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2CMS*SOV14B, D;26B,D;32B; 33B;34B;35B; 61A,B;63A,B	Primary Containment Isolation - CMS	B10c2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-
2CPS*AOV104, 105,110,111 *SOV119,120	Primary Containment Isolation - CPS	B10d1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2CPS*MOV106, 107,108,109 *SOV121,122	Primary Containment Isolation - CPS	B10d2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-
2CSH*AOV108 *MOV105,107 111,118 E33-F005 -F012,F004, F023,F015	Primary Containment Isolation - CSH	B10e	1	N/A	N/A	Yes	Yes	I	Div. 3	P601	-
2CSL*AOV101 *MOV104,112 E21-F006 -F005,F001	Primary Containment Isolation - CSL	B10f	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-
2DER*MOV120, 131	Primary Containment Isolation - DER	B10g1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2DEP*MOV119, 130	Primary Containment Isolation - DEP	B10q2.	1	N/A	N/A	Yes	Yes	I	Div. 2	P873	-
2DFR*MOV120, 139	Primary Containment Isolation - DFR	B10h1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2DFR*MOV121, 140	Primary Containment Isolation - DFR	B10h1	1	N/A	N/A	Yes	Yes	I	Div. 2	P873	-
2FPW*SOV218, 220	Primary Containment Isolation - FPW	B10j1	1	N/A	N/A	Yes	Yes	I	Div. 1	P849	36
2FPW*SOV219, 221	Primary Containment Isolation - FPW	B10j2	1	N/A	N/A	Yes	Yes	1	Div. 2	P849	36
2FWS*MOV21A, B 822-F065A,B	Primary Containment Isolation - FWS	B10k1	1	N/A	N/A	Yes	Yes	I	Div. 1	P603	-
2FWS*AOV23A, B 822-F032A,B	Primary Containment Isolation - FWS	B10k2.	1	N/A	N/A	Yes	No	II	Non-1E	P603	9
2HCS*MOV1A, 2A,3A,4A, 5A,6A	Primary Containment Isolation - HCS	B10L1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2HCS*MOV1B, 2B,3B,4B, 5B,6B	Primary Containment Isolation - HCS	B10L2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-

TABLE 421.36-1 (Cont)

SNEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2IAS*SOV164, 166,167,168	Primary Containment Isolation - IAS	B10m1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601/ P851	38
2IAS*SOV165, 180,184,185	Primary Containment Isolation - IAS	B10m2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601/ P851	38
2ISC*MOV121, 122,126,136 183,164 *AOV156,157 E51-F064,F068, F013, F031, F019, F080, E51-F065,F066	Primary Containment Isolation - ICS	B10n1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-
2ICS*MOV128, 148,170 E51-F063, F086,F076	Primary Containment Isolation - ICS	B10n2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601	-
2LMS*SOV153, 157	Primary Containment Isolation - LMS	B10p1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2LMS*SOV152, 156	Primary Containment Isolation - LMS	B10p2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-
2MSS*MOV112, 20B B22-F019,N/A	Primary Containment Isolation - MSS	B10q1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602	-
2MSS*HIV7A, B,C,D B22-F028A,B, C,D	Primary Containment Isolation - MSS	B10q2	1	N/A	N/A	Yes	Yes	I	BPS Div. 1	P602	-

TABLE 7A21.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2MSS*MOV111 B22-F016	Primary Containment Isolation - MSS	B10q3	1	N/A	N/A	Yes	Yes	I	Div. 2	P602	-
2MSS*HYV6A, B,C,D B22-F022A,B, C,D	Primary Containment Isolation - MSS	B10q4	1	N/A	N/A	Yes	Yes	I	RPS Div 2	P602	-
C51-J004A,B, C,D,E	Primary Containment Isolation - RMS	B10r	1	N/A	N/A	Yes	No	II	Non-1E	P607	10
2RCS*SOV65A, B;66A,B;67A, B 2RCS*SOV104 B35-F020	Primary Containment Isolation - PCS	B10s1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602	-
2RCS*SOV79A, B;80A,B;81A, B;82A,B 2RCS*SOV105 B35-F019	Primary Containment Isolation - RCS	B10s2	1	N/A	N/A	Yes	Yes	I	Div. 2	P602	-
2RHS*MOV1A, 15A,16A,24A, 25A,26A,27A, 30A,33A,39A, 40A,67A,104, 113 B12-F004A,F016A,F041A, F042A,F017A,F078A,F079A F105A,F027A,F050A,F053A F099A,F023,F008	Primary Containment Isolation - RHS	B10t1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-

TABLE 821.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2RHS*MOV18, 1C;15B;16BC; 24A,B;25B; 26B;27B;30B; 33B;34B;40B;61B;112 E12-F004B,C;F016B; F041BC;F042B,C;F017B;F074B; F073B;F105B;F037B;F050B; F059B;F099B;F009	Primary Containment Isolation - RHS	B10t2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601	-
2SAS*MCV160, 161	Primary Containment Isolation - SAS	B10u1		N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2SAS*MCV162, 163	Primary Containment Isolation - SAS	B10u2	1	N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2SLS*MOV5A C41-F006A	Primary Containment Isolation - SLS	B10v1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-
2SLS*MOV5B C41-F006B	Primary Containment Isolation - SLS	B10v2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601	-
2WCS*MOV112, 200A G33-F004, F040	Primary Containment Isolation - WCS	B10w1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602	-
2WCS*MOV102 G33-F001	Primary Containment Isolation - WCS	B10w2	1	N/A	N/A	Yes	Yes	I	Div. 2	P602	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NFD I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
	Radioactive Concentration in Primary Coolant	C1	1	Unit 1 H.P. Lab	(ltr)	(ltr)	(ltr)	II	Non-1E	-	4
	Analysis of Primary Coolant Gamma Spectrum	C2	3	Unit 1 H.P. Lab	(ltr)	(ltr)	(ltr)	II	Non-1E	-	4
N/A N/A	Core Temperature	C3	1	-	-	-	-	-	-	-	6
See Note 11 See Note 11	Reactor Coolant System Pressure	C4	1	-	-	-	-	-	-	-	11
See Note 12	Primary Containment Area Radiation	C5	3	-	-	-	-	-	-	-	12
N/A N/A	Drywell Drain Sumps Level	C6	1	-	-	-	-	-	-	-	13
2CMS*LT11A	Suppression Pool Water Level (Narrow Rng)	C7a	1	Rx Bldg (Sec Contmt)	197-202 ft	Yes	Yes	I	Div. 1	P601	14
2CMS*LT11B	Suppression Pool Water Level (Narrow Rng)	C7b	1	Rx Bldg (Sec Contmt)	197-202 ft	Yes	Yes	I	Div. 2	P898	14

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CMS*LT9A	Suppression Pool Water Level (Wide Rng)	C7c	1	Rx Bldg (Sec Contmt)	192-217 ft	Yes	Yes	I	Div. 1	P601	14
2CMS*LT9B	Suppression Pool Water Level (Wide Rng)	C7d	1	Rx Bldg (Sec Contmt)	192-217 ft	Yes	Yes	I	Div. 2	P898	14
See Note 15	Drywell Pressure	C8	1	-	-	-	-	-	-	-	15
See Note 16	Reactor Coolant System Pressure	C9	1	-	-	-	-	-	-	-	16
See Note 17	Primary Containment Pressure	C10	1	-	-	-	-	-	-	-	17
2CMS*AIT6A	Containment Hydrogen Concentration - A	C11a	1	Rx Bldg North Aux Bay	0-30%	Yes	Yes	I	Div. 1	P601	-
2CMS*AIT6B	Containment Hydrogen Concentration - B	C11b	1	Rx Bldg South Aux Bay	0-30%	Yes	Yes	I	Div. 2	P898	-
2CMS*AIT71A	Containment Oxygen Concentration - A	C12a	1	Rx Bldg North Aux Bay	0-10%	Yes	Yes	1	Div. 1	P601	-
2CMS*AIT71B	Containment Oxygen Concentration - B	C12b	1	Rx Bldg North Aux Bay	0-10%	Yes	Yes	I	Div. 2	P898	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2EMS-CAB170	Containment Effluent Radioactivity	C13	3	Main Stack Enclosure	Isotopic 10^{-7}-10^{5} uci/cc	No	Yes	II	Non-1E	P882	39
2RMS-CAB180 -	Effluent Radioactivity	C14	2	Turb Bldg Turb Oper Floor	Isotopic 10^{-7}-10^{5} uci/cc	No	Yes	II	Non-1E (UPS)	P882	39
2FMS-FT1A,B C33-N001A,B	Main Feedwater Flow -A,B		3	Turb Bldg	Turb Bldg 0-8.5 lbs/hr (each)	No	No	II	Non-1E	P603	
2CMS-LT8A,B	Condensate Storage Tk Level - A, B	D2	3	Cond Stor TK1A,TK1B	0-500 K gal (each)	No	No	II	Non-1E	P851	-
2RHS*PT64A	Suppression Chamber Spray Header Flow - A	D3a	2	Rx Bldg (Sec Contmt)	0-450 gpm	Yes	Yes	I	Div. 1	P601	-
2RHS*PT64B	Suppression Chamber Spray Header Flow - B	D3b	2	Rx Bldg (Sec Contmt)	0-450 gpm	Yes	Yes	I	Div. 2	P601	-
See Note 18	Drywell Pressure	D4	2	-	-	-	-	-	-	-	18
See Note 19	Suppression Water Level (Weir Well)	D5	2	-	-	-	-	-	-	-	19
2CMS*TE67A, 68A,69A,70A	Suppression Pool Water Temp-A	D6a	2	Suppression Pool	50-250°F	Yes	Yes	I	Div. 1	P601	20
2CMS*TE67B, 68B,69B,70B	Suppression Pool Water Temp-B	D6b	2	Suppression Pool	50-250°F	Yes	Yes	I	Div. 2	P601/ P598	20

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CMS*TE101 thru & inc 109	Drywell Atmos Temp - A	D7a	2	Drywell	0-400°F	Yes	Yes	I	Div. 1	P873	20
2CMS*TE116 thru & inc 124	Drywell Atmos Temp - B	D7b	2	Drywell	0-400°F	Yes	Yes	I	Div. 2	P875	20
2RHS*FT63A	Drywell Spray Header Flow - A	D8a	2	Rx Bldg (Sec Contmt)	0-7,950 gpm	Yes	Yes	I	Div. 1	P601	-
2RHS*FT63B	Drywell Spray Header Flow - B	D8b	2	Rx Bldg (Sec Contmt)	0-7,950 gpm	Yes	Yes	I	Div. 2	P601	-
See Note 19	Main Steam Line Isolation Valve Leakage Control System Press	D9	2	-	-	-	-	-	-	-	19
See Note 19											
2SVV*ZT220 237	Primary Safety Relief Valve Position	D10a	2	Acoustic Sensor on Tail Pipe (18 total)	-	Yes	Yes	I	Div. 1	*SVV-PNL 140	-
2IAS*PT181	Primary Safety Relief Valve ADS Header Pressure-A	D10b	2	Rx Bldg (Sec Contmt)	0-250 psig	Yes	Yes	I	Div. 1	P601	-
2IAS*PT230, 231,232	Primary Safety Relief Valve - ADS Tank Pressure	D10c	2	Rx Bldg (Sec Contmt)	0-200 psig	Yes	Yes	I	Div. 1	P601	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2IAS*PT186	Primary Safety Relief Valve - ADS Header Pressure-B	D10d	2	Rx Bldg (Sec Contmt)	0-250 psig	Yes	Yes	I	Div. 2	P601	-
2IAS*PT233, 234,235,236	Primary Safety Relief Valve - ADS Tank Pressure	D10e	2	Rx Bldg (Sec Contmt)	0-200 psig	Yes	Yes	I	Div. 2	P601	-
See Note 19	Isolation Condenser - Shell Side Water Level	D11	2	-	-	-	-	-	-	-	19
See Note 19											
See Note 19	Isolation Condenser Valve Position	D12	2	-	-	-	-	-	-	-	19
See Note 19											
2ICS*FT102 E51-N051	RCIC Flow	D13	2	Rx Bldg (Sec Contmt)	0-800 gpm	Yes	Yes	I	Div. 1	N/A	-
2CSH*FT105 E22-N056	HPCI/S Flow	D14	2	Rx Bldg (Sec Contmt)	0-1000 gpm	Yes	Yes	I	Div. 3	N/A	21
2CSL*FT107 E21-N051	LPCS Flow	D15	2	Rx Bldg (Sec Contmt)	0-10,000 gpm	Yes	Yes	I	Div. 1	N/A	-
2RHS*FT14A E12-N015A	LPCI Flow	D16a	2	Rx Bldg (Sec Contmt)	0-8,400 gpm	Yes	Yes	I	Div. 1	P601	-
2RHS*FT14B,C E12-N015B,C	LPCI Flow	D16b	2	Rx Bldg (Sec Contmt)	0-8,400 gpm	Yes	Yes	I	Div. 2	P601	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2SLS*FT113 E81-N007	SLCS Flow	D17	2	Rx Bldg (Sec (Contmt)	0-86 gpm	Yes	Yes	I	Div. 1	P601	-
2SLS*LT103 C41-N001	SLCS Storage Tank Level	D18	2	Rx Bldg (Sec Contmt)	0-10,000 Gal	Yes	Yes	I	Div. 1	P601	-
See Note 22 See Note 22	RHR System Flow	D19	2	-	-	-	-	-	-	-	22
2PHS*TE13A E12-N027A	RHR Heat Exchanger Outlet Temp - A	D20a	2	Rx Bldg (Sec Contmt)	0-600°F	No	No	II	Non-1E	P601	-
2PHS*TE13B E12-N027B	RHR Heat Exchanger Outlet Temp - B	D20b	2	Rx Bldg (Sec Contmt)	0-600°F	No	No	II	Non-1E	P601	-
2SWP*TE31A	Cooling Water Temp to ESF System Components - A	D21a	2	Screen-well Bldg	35-130°F	Yes	Yes	I	Div 1	P601	20
2SWP*TE31B	Cooling Water Temp to ESF System Components - B	D21b	2	Screen-well Bldg	35-130°F	Yes	Yes	I	Div. 2	P601	20
2SWP*FT13A E12-N007A	Cooling Water Flow to ESF System Components - A	D22a	2	Rx Bldg (Sec Contmt)	0-10,000 gpm	Yes	Yes	I	Div. 1	P601	23
2SWP*FT13B E12-N007B	Cooling Water Flow to ESF System Components - B	D22b	2	Rx Bldg (Sec Contmt)	0-10,000 gpm	Yes	Yes	I	Div. 2	P601	23

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2SWP*PT76A	Cooling Water Flow to ESF System Components - Div. 1 psi	D22c	2	Diesel Gen. Bldg	0-860 gpm	Yes	Yes	I	Div. 1	P852	24
2SWP*PT6B	Cooling Water Flow to ESF System Components - Div. 2 DSL	D22d	2	Diesel Gen. Bldg	0-860 gpm	Yes	Yes	I	Div. 2	P852	24
2SWP*PT535	Cooling Water Flow to ESF System Components - Div. 3 DSI	D22e	2	Diesel Gen. Bldg	0-650 gpm	Yes	Yes	I	Div. 2	P852	24
2LWS-2A,B,C; 26A,B;276; 280	High Radio-activity Liquid Tank Level	D23	3	Radwaste Bldg	0-100%	No	No	II	Non-1E	LWCS Computer Graphics	-
2HVP*AOD1A, 6A,9A,10A	Emergency Ventilation Damper Position	D24a	2	Rx Bldg (Sec Contmt)	N/A	Yes	Yes	I	Div. 1	P870	-
2HVP*AOD1B, 6B,9B,10B	Emergency Ventilation Damper Position	D24b	2	Rx Bldg (Sec Contmt)	N/A	Yes	Yes	I	Div. 2	P871	-
N/P	Status of Stdby Pwr Sources - Battery Voltage - 1	D25a	2	2BYS* SWG002A	0-150 vdc	Yes	Yes	I	Div. 1	P852	-
2BYS*E/E1A	Status of Stdy Pwr Sources - Battery Current - 1	D25b	2	2BYS* SWG003A	-2000 to 3000 amps	Yes	Yes	I	Div. 1	P852	-

TABLE 421.36-1 (Cont)

SNEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
N/R	Status of Stdby Pwr Sources - Battery Voltage - 2	D25c	2	2BYS* SWG003B	0-150 vdc	Yes	Yes	I	Div. 2	P852	-
2BYS*E/E18	Status of Stdby Pwr Sources - Battery Current - 2	D25d	2	2BYS* SWG002B	-2000 to +2000 amps	Yes	Yes	I	Div. 2	P852	-
N/R	Status of Stdby Pwr Sources - Battery Voltage - 3	D25e	2	2EGS* PNL002	0-150 vdc	Yes	Yes	I	Div. 3	P852	-
2BYS*E/E101	Status of Stdby Pwr Sources - Battery Current - 3	D25f	2	2EGS* PNL002	-100 to +100 amps	Yes	Yes	I	Div. 3	P852	-
N/R	Status of Stdby Pwr Sources - UPS Voltage - A	D25g	2	2YBS* UPS2A	0 to 120 vac	Yes	Yes	I	Div. 1	PP52	-
N/R	Status of Stdby Pwr Sources - UPS Current - A	D25h	2	2YBS* UPS2A	0 to 250 amps	Yes	Yes	I	Div. 1	N/A	-
N/R	Status of Stdby Pwr Sources - UPS Voltage - B	D25j	2	2UBS* UPS2B	0 to 120 vac	Yes	Yes	I	Div. 2	P852	-
N/R	Status of Stdby Pwr	D25k	2	2UBS* UPS1B	0 to 250 amps	Yes	Yes	I	Div. 2	N/A	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Note
	Sources - UPS Current - B										
N/R	Status of Stdby Pwr Sources - 600V Swgr Voltage	D25l	2	2EJS*AS1	0 to 750 vac	Yes	Yes	I	Div. 1	2EJS*US1	25
N/R	Status of Stdby Pwr Sources - 600V Swgr Current	D25m	2	2EJS* X1A,B	0 to 3,000 amps	Yes	Yes	I	Div. 1	N/A	26
N/R	Status of Stdby Pwr Sources - 600V Swgr Voltage	D25n	2	2EJS* US3	0 to 750 vac	Yes	Yes	I	Div. 2	2EJS*US3	25
N/R	Status of Stdby Pwr Sources - 600V Swgr Current	D25p	2	2EJS* X3A,B	0 to 3,000 amps	Yes	Yes	I	Div. 2	N/A	26
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Voltage	D25q	2	Swgr	0 to 4.16 kv	Yes	Yes	I	Div. 1	P852	25
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Current	D25r	2	Swgr	0-1000 amp Diesel feed or 1,500 amp normal and alt feeds	Yes	Yes	I	Div. 1	P852	27

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Voltage	D25s	2	Swgr	0 to 4.16 kv	Yes	Yes	I	Div. 2	P852	25
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Current	D25t	2	Swgr	0-1,000 amp Diesel Feed 0-1,500 amp normal & alt feeds	Yes	Yes	I	Div. 2	P852	27
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Voltage	D25u	2	Swgr	0 to 4.16 kv	Yes	Yes	I	Div. 3	P852	25
N/R	Status of Stdby Pwr Sources - 4kV Swgr Current	D25v	2	Swgr	0-600 amp Diesel Feed 0-1,500 amp normal & alt feeds	Yes	Yes	I	Div. 3	P852	27
See Note 28	Status of Stdby Pwr Sources - Air for ADS	D25w	2	-	-	-	-	-	-	-	28
2RMS*RE1A,C	Primary Containment Area Radiation High Rng	E1a	1	Drywell	1-10^7 R/hr	Yes	Yes	I	Div. 1	P880	-
2RMS*RE1B,D	Primary Containment Area Radiation High Rng	E1b	1	Drywell	1-10^7 R/hr	Yes	Yes	I	Div. 2	P880	-
See Note 29	Secondary Containment Area Radiation	E2	2	-	-	-	-	-	-	-	29
See Note 30	Vital Area Radiation	E3	3	See Note 30	See Note 30	NO	No	II	Non-1E	Radn Computer	30,31

TABLE 821.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
	Monitors									Graphics	
See Note 32	Effluent Radiation Released from Plant	E4	2	-	-	-	-	-	-	-	32
See Note 33	Environs Radiation and Radio-activity	E5	3	-	-	-	-	-	-	-	31,33
See Note 34	Meteorology	E6	3	-	-	-	-	-	Non-1E	P882	31,34
2SSP-IPNL101 D24-D007	Accident Sampling Primary Coolant and Containment Air	E7	3	Radwaste Sample Room	N/A	No	No	II	Non-1E	-	4,35

07-464-91B

JAN 23 1989

Docket Nos. 50-220
50-410

Niagara Mohawk Power Corporation
ATTN: Mr. C. V. Mangan
Senior Vice President
c/o Miss Catherine R. Seibert
301 Plainfield Road
Syracuse, New York 13212

Gentlemen:

Subject: Inspection Report Nos. 50-220/88-34 and 50-410/88-32

This refers to the routine safety inspection conducted by R. J. Paolino of this office on November 14-18, 1988, of activities at Niagara Mohawk Power Corporation Units 1&2 Nuclear Power Plants, located in Scriba, New York. These activities are authorized by NRC Operating License Nos. DPR-63/NPF-69. Discussions of our findings were held by Mr. Paolino with Mr. J. Willis and other members of your staff at the conclusion of the inspection.

Specific areas examined during this inspection included your compliance with the order issued to implement R.G. 1.97 and are described in the NRC Region I Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Based on the results of this inspection, it appears that several of your activities were not conducted in full compliance with NRC requirements, as set forth in the Notice of Violation, enclosed herewith as Appendix A. The Notice of Violation has been categorized by severity level in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (Enforcement Policy). You are required to respond to this letter and in preparing your response, you should follow the instructions in Appendix A. In addition to this violation, a number of apparent deviations were noted with regard to Unit 1 post accident monitoring equipment and the guidance provided in Regulatory Guide 1.97 (Rev. 2). No technical basis was provided to the NRC to support these apparent deviations. During a post inspection meeting with the NRC on December 20, 1988 your staff stated that a review was in process to identify Unit 1 deviations from R.G. 1.97. The results of this review are to be discussed with the NRC in a meeting scheduled for February 3, 1989. It is important that management attention be given to completing this review in a timely manner.

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The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Your cooperation with us in this matter is appreciated.

Sincerely,

Original Signed By:
Thomas T. Martin

Thomas T. Martin, Director
Division of Reactor Safety

Enclosures:

1. Appendix A, Notice of Violation
2. Combined NRC Region I Inspection Report Number 50-220/88-34 and 50-410/88-32

cc w/encl:
J. A. Perry, Vice President, Quality Assurance
L. Burkehart, Vice President, Nuclear Generation
W. Hansen, Manager Corporate Quality Assurance
K. Dahlberg, Unit 1 Station Superintendent
R. Randall, Unit 1 Superintendent, Operations
C. Beckham, Manager Nuclear Quality Assurance Operations
W. Drews, Technical Superintendent
J. Willis, General Station Superintendent
C. Terry, Vice President Nuclear Engineering and Licensing
J. F. Warden, New York Consumer Protection Branch
Troy B. Conner, Jr. Esquire
John W. Keib, Esquire
Director, Power Division
State of New York, Department of Law
Licensing Project Manager, NRR
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New York

JAN 23 1989

Niagara Mohawk Power Corporation 3

bcc w/encl:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
DRP Section Chief
PAO (2) SALP Reports and All Inspection Reports
Robert J. Bores, DRSS
B. Clayton, EDO

RI:DRS	RI:DRS	RI:DRS
Paolino/tlm/rw	Anderson	Durr
1/10/89	1/10 /89	1/23/89

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APPENDIX A

NOTICE OF VIOLATION

Niagara Mohawk Power Corporation
Nine Mile Point Unit 1 Nuclear Power Plant

Docket No. 50-220
License No. DPR-63

As a result of the inspection conducted on November 14-18, 1988, and in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (Enforcement Policy) (1988), the following violation has been identified:

10 CFR 50.49 paragraph (b)(3) and (f) requires that certain post accident monitoring equipment be qualified and qualification of each item of electrical equipment be based on testing or experience with identical equipment or with similar equipment with supporting analysis to show that the equipment is acceptable; 10 CFR 50.49, paragraph (g) requires that each item of electrical equipment important to safety be qualified and that qualification must be completed at a time no later than November 30, 1985.

Section 1.31 "Design and Qualification Criteria" - Category 1 of Regulatory Guide 1.97, Rev. 2 states in part: The instrumentation should be qualified in accordance with Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," and the methodology described in NUREG-0588, "Interim Staff Position on EQ of Safety-Related Electrical Equipment". Qualification applies to the complete instrumentation channel from sensor to display.

Contrary to the above, on November 18, 1988 the following Category 1, Regulatory Guide 1.97 instruments did not have sufficient documentation to establish their Environmental Qualification in accordance with 10 CFR 50.49 requirements:

1. Suppression Pool Water Level, Category 1, Type A variable (Downcomer submergence), Instrument No. 58-04.

2. Drywell Atmosphere Temperature, Category 1, Type A variable, Thermocouple Numbers 201-36A, 201-50A and 201-51A.

3. Suppression Pool Water Temperature, Category 1, Type A variable, Sensing Element Numbers TE201.2-491, TE201.2-492, TE201.2-517 and TE201.2-518.

This is a Severity Level IV violation. (Supplement I)

Pursuant to the provisions of 10 CFR 2.201, Niagara Mohawk Power Corporation is hereby required to submit to this office within thirty days of the date of the letter which transmitted this Notice, a written statement or explanation in reply, including: (1) the corrective steps which have been taken and the results achieved; (2) corrective steps which will be taken to avoid further violations; and (3) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending this response time.

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos. 50-220/88-34
50-410/88-32

Docket Nos. 50-220/50-410

License Nos. DPR-63/NPF-69 Priority - Category C

Licensee: Niagara Mohawk Power Company
301 Plainfield Road
Syracuse, New York 13212

Facility Name: Nine Mile Point Nuclear Station - Units 1 and 2

Inspection At: Scriba, New York

Inspection Conducted: November 14-18, 1988

Inspectors: R. J. Paolino, Senior Reactor Engineer/PSS — date 1-10-89

R. K. Mathew, Reactor Engineer/PSS — date

Other Participants and Contributors to the Report Include:

Allan C Udy, NRC Contractor - INEL

Ron VanderBeek, NRC Contractor - INEL

Approved by: C. J. Anderson, Chief - Plant Systems Section - EB/DRS — date 1/10/89

Inspection Summary: Inspection on November 14-18, 1988 (Combined Inspection Reports Nos. 50-220/88-34 and 50-410/88-32)

Areas Inspected: A special announced inspection of representative subsystems of the post-accident monitoring instrumentation in accordance with Regulatory Guide 1.97, Revision 2, for Unit 1 and Revision 3 for Unit 2. The inspection assessed the licensee conformance to requirements specified in the Order, dated June 12, 1984, to commitments made per generic letter 82-33 and supplement 1 to NUREG-0737.

Results: Of the areas inspected there was one apparent violation with three examples of not complying with 10 CFR 50.49 requirements for safety related electrical equipment at NMP-1. In addition, significant deficiencies were identified by the licensee and the NRC. These deficiencies indicate that the licensee does not conform to Section 6 of Supplement 1 to NUREG-0737 regarding Regulatory Guide 1.97 "Applications to Emergency Response Facilities." The deficiencies related to the following areas:

- Redundancy - A number of deficiencies were identified with control room R.G. 1.97 instrumentation regarding electrical and physical separation.

- Interfaces - A number of circuits were identified which either lacked 1E/non-1E isolation devices or which contained inadequate isolation devices.

- Tagging & Marking - R.G. 1.97 instrumentation in the control room was not appropriately marked.

- Display & Recording - Recording of instrumentation readout for one channel for certain instruments was not provided.

Specific findings for NMP-1 are identified in attachment 1 (A&B).

For NMP-2 minor deficiencies were identified involving differences in documented instrument ranges and installed instrument configuration. The majority of these deficiencies require changes to the FSAR and engineering drawings. Specific deficiencies for NMP-2 are identified in attachment 1(C).

DETAILS

1.0 Persons Contacted

1.1 Niagara Mohawk Power Corporation (NMPC)

* C. G. Beckham	QA/QC
* J. L. Benson	Nuclear Consultant
* U. Buiva	Unit 2 Engineering
T. J. Chwalek	Chem & Rad. Mgt, Unit 1
* F. J. Constance	Nuclear Electrical Design
* K. A. Dahlberg	Stations Manager, Operations, Unit 1
* J. L. Dillon	Supervisor QA Audits
W. B. Davey	NMP2 Oper
E. Dehart	Engineering, Unit 2
J. N. Duell	Chem & Rad Mgt, Unit 1
* W. P. James, Jr.	I&C, Unit 1
J. Jirousek	EQ Manager
J. L. Kibbe	NMP2 Oper
* J. R. Kinsley	I&C Supervisor
T. Kolceski	EQ Engineering
* E. W. Leach	Radiation Protection
* D. Lesurdo	PSC
* P. D. MacEwan	NYSEG
* G. H. Montgomery	Regulatory Compliance
S. C. Nicolos	I&C, Unit 2
* R. J. Pasternak	Manager, Site Engineering
L. Price	EQ Engineer
* N. L. Rademacher	Director of Compliance, Unit 1
* B. Randall	Operations, Superintendent, Unit 1
* A. D. Sassani	Project Engineering, Unit 2
* K. J. Sweet	Maintenance Superintendent, Electr, Unit 1
* J. H. Snyder	Site Liasion Engineering
K. B. Thomas	Consulting Services
* P. Volza	Rad Protection Manager
* S. W. Wilczek, Jr.	Manager Nuclear Technology
* J. L. Willis	General Superintendent
* K. Yackel	I&C, Unit 1

1.2 Stone and Webster Engineering Corporation

A. Issa	Seismic Engineer
L. Illy	EQ Engineer
A. Sclocchini	Design Engineer

1.3 U.S. Nuclear Regulatory Commission

* C. Anderson	Chief, Plant Systems Section EB/DRS
* R. Temps	Resident Inspector

* denotes personnel attending exit meeting of November 18, 1988

2.0 Introduction

Background

The purpose of this inspection was to verify the implementation of instrumentation systems for assessing plant conditions during and following the course of an accident that meets the criteria specified in Regulatory Guide (RG) 1.97, Revision 2 for Unit 1 and Revision 3 for Unit 2. These systems were inspected to determine if they were installed in accordance with generic letter number 82-33 "Requirements for Emergency Response Capability" (Supplement 1 to NUREG-0737). This letter, issued on December 17, 1982, specifies those requirements regarding emergency response capabilities that have been approved by the NRC for implementation. This supplement also discusses, in part, the application of RG 1.97 to the emergency response facilities, including the control room (CR), technical support center (TSC) and the emergency response facility (EOF) at power plants. Regulatory Guide 1.97 identifies the plant variables to be measured and the instrumentation criteria for assuring acceptable emergency response capabilities during and following the course of an accident.

Regulatory Guide 1.97 divides Post Accident Instrumentation into 3 categories and 5 types. The three design categories are noted as 1, 2 and 3. Category 1 has the most stringent design requirements and category 3 has the least stringent. The five types of instrumentation identified in the Regulatory Guide are types A, B, C, D, and E. Type A variables are plant specific and classified by the licensee. Type B variables provide information to indicate that plant safety functions are being accomplished. Type C variables provide information regarding the breach of barriers for fission product release. Type D variables indicate the operation of individual safety systems. Type E variables are those that indicate and determine the magnitude of the release of radioactive materials. Each variable type can be any design category. However, type A variables can only be design Category 1.

Correspondence

The licensee's response to RG 1.97 for Unit 1 was provided in submittals dated April 2, 1984; October 18, 1985 and December 6, 1985. For Unit 2 the licensee's RG 1.97 response was provided in submittals dated October 5, 1984, January 20, 1986 and March 1986. The Safety Evaluation Report (SER) was issued by the NRC on March 3, 1986 for Unit 1 and May 5, 1986 for Unit 2. The SERs specify staff positions for licensee deviations and exceptions from the guidelines.

References

The specific references used to assess the licensee's response to Regulatory Guide 1.97 are as identified below:

- Regulatory Guide 1.97, Revision 2 and Revision 3 "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."
- Safety Evaluation Report - Emergency Response Capability, Conformance to Regulatory Guide 1.97, Rev. 2 & 3.
- Niagara Mohawk Power Corporation Units 1&2 Final Safety Analysis Report (FSAR), Chapter 7.
- Licensee procedures and reference drawings as shown in attachments 1 & 2.
- Technical Supplement to Petition for Conversion from Provisional Operating Licenses to Full Term Operating License, Nine Mile Nuclear Station, July 1972.
- Amendment 1 to Application to Convert Provisional Operating License to Full Term Operating License, November 1973.

Inspection Scope

The NRC inspection scope included: equipment qualification (Seismic and Environmental), redundancy of power supplies, measured variables, display and recording methods used, independence and separation of electrical circuits, range and overlapping features of multiple instrument indicators, equipment identification for RG 1.97 instruments, service, test and surveillance frequency, direct and indirect measurements of parameters of interest.

The safety related (Q) and EQ master equipment lists were reviewed for the instruments selected, to ascertain whether they had been evaluated and tested to the appropriate environmental, quality assurance (QA) and seismic qualification requirements. The QA procurement of these instruments was also reviewed.

3.0 Inspection Details

The inspectors held discussions with various members of the licensee's staff, reviewed drawings (Attachment 2) and procedures, and selected variables for systems walkdown. Walkdowns were performed for control room instruments to assess the implementation of the RG 1.97 Rev. 2 for Unit 1 and Rev. 3 for Unit 2.

Instrument variables reviewed for NMP-1 included reactor coolant level, reactor pressure, drywell pressure, drywell atmospheric temperature, containment hydrogen/oxygen concentration, suppression pool water level and suppression pool water temperature. The NMP-2 inspection included primary containment pressure, coolant water flow to ESF system component, coolant water temperature to ESF system component, RHR heat exchanger outlet temperature, LPCI system flow, drywell spray flow, suppression pool pressure and primary containment area radiation.

Characteristics examined for each variable include identity, location, function, separation (physical/electrical), isolation, seismic, power source, environmental qualification status and instrument range. Items not conforming with Section 6 of Supplement 1 to NUREG-0737 are discussed in the following section.

4.0 Regulatory Guide 1.97 Variables Evaluated (Unit 1)

4.1 Reactor Pressure

Regulatory Guide 1.97 classifies this as a Category 1 variable and the licensee's commitment to Section 6, Supplement 1 of NUREG-0737 resubmittal dated October 5, 1987 has specified that this variable is a Type A variable. The monitoring of this variable is accomplished by using two pressure sensing channels with indication in the control room. One channel has a recorder. The variable is also monitored by the Safety Parameter Display System (SPDS). The instrumentation for this variable does not meet the Category 1 criteria specified in Section 6, Supplement 1 to NUREG-0737 regarding Regulatory Guide 1.97 requirements for separation. Section 1.3.1.b of the Regulatory Guide states, in part, that redundant or diverse channels should be electrically independent and physically separated from each other. The pressure recorder PR-1075 interfaces with PT36-31 and PT36-32 through switch no. 1D20A. No isolation device or separation is provided between channels. Also, all power sources, numbers ACV-P, ACV-R, and ACV-S normally are powered by RPS Bus 11, Circuit 12. Therefore, they are not considered independent, redundant power sources. This apparent deviation from Section 6 of Supplement 1 to NUREG-0737 requirements regarding Regulatory Guide 1.97 is an unresolved item pending the licensee developing the technical basis for the variations from R.G. 1.97 and NRC review of this information. (88-220/88-34-01)

4.2 Drywell Atmospheric Temperature

Regulatory Guide 1.97 generally classifies the drywell atmospheric temperature as a Category 2 variable. However, the licensee has specified that this variable is a Type A variable. Because it is a Type A variable, the instrumentation supplied should be Category 1. The monitoring of this variable is accomplished by three channels of instrumentation. The instrumentation for this variable does not meet the Category 1 guidance specified in RG 1.97. Section 1.3.1(b) of

the Regulatory Guide states, in part, that: "redundant or diverse channels should be electrically independent and physically separated from each other... At least one channel should be displayed on a direct indicating or recording device." All of these instrument channels (Nos. TE201-36, TE201-27 and TE201-33) have a common power supply. Therefore, the channels were not provided with complete independence and redundancy. No recorder for readout data is provided. This apparant deviation from Section 6 of Supplement 1 to NUREG-0737 regarding compliance with Regulatory Guide 1.97 criteria is an unresolved item pending the licensee developing the technical basis for the variations from R.G. 1.97 and NRC review of this information. (50-220/88-34-02)

In addition, environmental qualification for the three instrument channel temperature sensing elements (TE201-36A, TE-201-50A and TE201-51A) has not been established. This is a violation of 10 CFR 50.49 (b)(3) and (F) which requires that these items of electrical equipment be qualified. (50-220/88-34-03)

4.3 Suppression Pool Water Level

Regulatory Guide 1.97 classifies suppression pool water level as a Category 1 variable. The licensee's resubmittal commitment to Section 6 of Supplement 1 to NUREG-0737 dated October 5, 1987 has specified that this variable is a Type A variable. The monitoring of this variable is accomplished by three instrument channels (LT58-04, LT58-05 & LT58-06). The instrumentation for this variable does not fully meet the Category 1 criteria specified in RG 1.97. The level transmitter LT58-04 was not EQ listed and environmental qualification of this item has not been established. This is in violation of 10 CFR 50.49(b)(3) and (F) which requires that each item of electrical equipment important to safety be qualified. (50-220/88-34-04)

4.4 Suppression Pool Water Temperature

Regulatory Guide 1.97 classifies suppression pool water temperature as a Category 2 variable. However, the licensee has specified that this variable is a Type A variable. Therefore, the instrumentation should conform to Category 1 criteria. The monitoring of this variable is accomplished by two channels of instrumentation (TT201.2-517 and TT201.2-518). The instrumentation for this variable does not fully meet the Category 1 criteria specified in RG 1.97. The thermocouples (TE201.2-491, TE201.2-492) and transmitters (TT201.2-517, TT201.2-518) did not have supporting documentation to verify that they are environmentally qualified. This item is in violation of 10 CFR 50.49(b)(3) and (F) which requires that each item of electrical equipment important to safety be qualified. (50-220/88-34-05)

4.5 Isolation Devices

Where a Category 1 signal is used as input to a non-Category 1 system, Regulatory Guide 1.97 specifies the use of isolation devices that are fully qualified for use in Category 1 circuits. The inspectors examined interconnecting diagrams for torus level (3), drywell pressure (4) and reactor coolant (5) instrumentation circuits. Six out of twelve circuits examined did not use any isolation devices. The six instrument channels include: level transmitter nos. LT58-04, LT58-05 and LT58-06, pressure transmitter nos. PT201.2-483, PT201.2-484 and reactor coolant level transmitter LT36-33. Typically, the inspectors found that either there were no isolation devices or that signals are directed to the plant computer by way of a dropping resistor, and in some cases, medium impedance line input resistors. This does not conform to the Regulatory Guide 1.97, Category 1 criteria for isolation. This apparent deviation from the Regulatory Guide 1.97 criteria is an unresolved item pending the licensee developing the technical basis for variations from R.G. 1.97 and NRC review of this information. (50-220/88-34-06)

One isolation device (RIS-SC326) was identified for which documentation was not available to establish performance characteristics or seismic requirements. This item is unresolved pending NRC review of licensee evaluation and corrective action. (50-220/88-34-07)

5.0 Regulatory Guide 1.97 Variables Evaluated (Unit 2)

5.1 Reactor Coolant Level

Regulatory Guide 1 97 classifies reactor coolant level as a Category 1 variable. The licensee has specified that this variable is a Type A variable. The monitoring of this variable is accomplished by two channels of instrumentation (2ISC*LT13A & 2ISC*LT13B) for wide range level and two channels of instrumentation (2ISC*LT9A & 2ISC*LT9B) for the fuel zone level. In addition to providing indicators in the reactor control room, the variable is recorded and displayed on demand on the Safety Parameter Display System (SPDS).

During the inspection of the instrumentation (2ISC*LI13A & 2ISC*LI13B) in the control room for this variable, the range of the variable was found to be -165 to +35 inches of water, whereas the range specified by the FSAR is 230.69 to 430.69 inches of water. The licensee reviewed the design and determined that the range of 200" of water (-165" to +35") is required. The documentation will be changed to reflect actual field conditions. Other than the documentation correction, the instrumentation for this variable meets the criteria of RG 1.97 for Category 1 instrumentation and is acceptable.

This item is unresolved pending licensee document changes to reflect actual field conditions. (50-410/88-32-01)

5.2 Cooling Water Flow to Engineered Safety Features (ESF) System Components

Regulatory Guide 1.97 classifies cooling water flow to ESF system components as a Category 2 variable. Since it is a Category 2 variable, the instrumentation does not require seismic qualification. The monitoring of this variable is accomplished by two channels of instrumentation. The inspection for this variable noted that the range of the 2SWP*FI13A and 2SWP*FI13B channels is 0 to 8,000 gpm whereas the FSAR specifies 0 to 10,000 gpm. For the 2SWP*FI76A and B channels it was noted that the range is 0-1,400 gpm, whereas the FSAR specifies 0-860 gpm. It was also determined that the range for the 2SWP*FI535 channel should have been 0 to 1,000 gpm. The licensee plans to submit an FSAR change to reflect the correct instrument range.

In addition to revising the FSAR to correct the difference in range, the licensee must determine whether the range of 0 to 8,000 gpm for the 2SWP*FI13A and 2SWP*FI13B channel meets or exceeds the 0 to 110 percent design flow range recommended by RG 1.97. The above items are unresolved. (50-410/88-32-02)

With the exception of the range deficiencies, this instrumentation meets the Category 2 recommendations specified in RG 1.97 and is acceptable.

5.3 Low Pressure Coolant Injection (LPCI) System Flow

Regulatory Guide 1.97 classifies LPCI system flow as a Category 2 variable. Since it is a Category 2 variable, the instrumentation does not require seismic qualification. The monitoring of this variable is accomplished by three channels of instrumentation. The inspection of the instrumentation (2SWP*FI14A, 2RHS*FI14B & C) for this variable noted that the indication range is 0 to 10,000 gpm; whereas, the FSAR and the FSAR question for this variable indicates ranges of 0 to 8,400 gpm and 0 to 8,000 gpm respectively. The licensee has committed to correct the differences in the ranges specified in the FSAR and will determine whether the 0 to 10,000 gpm range is the recommended zero to 110 percent design flow range specified by RG 1.97. With the exception of the determination of zero to 110 percent design flow for this variable, the instrumentation meets the recommendations of RG 1.97 and is acceptable.

This item is unresolved pending licensee amendment of the FSAR and evaluation and determination on meeting 0-110 percent design flow range. (50-410/88-32-03)

5.4 Drywell Spray Flow

Regulatory Guide 1.97 classifies drywell spray flow as a Category 2 variable. Being a Category 2 variable, the instrumentation does not require seismic qualification. The monitoring of this variable is accomplished by two channels of instrumentation (2RHS*FT63A & 2RHS*FT63B). The inspection of the instrumentation (2RHS*FI63A & 2RHS*FI63B) for this variable noted that the indication range is 0 to 8,000 gpm; whereas, the FSAR specifies a range of 0 to 7950 gpm. The licensee has committed to correct the difference noted. The instrumentation meets the recommendations of RG 1.97 and is acceptable.

6.0 Physical Walkdown

The inspectors examined the Unit 1 and 2 control rooms to determine agreement with Regulatory Guide 1.97 guidance for electrical/physical separation, identification and adequacy of the instrumentation calibration program. For Unit 2, post-accident monitoring instruments designated as Type A, B and C and Categories 1 and 2 were specifically identified on the control panel (red outline) so that the operator would easily discern the intended use under accident conditions. Location and accessibility provides easy access for administrative control of all set point adjustments, module calibration and test points. The inspectors determined that the Unit 2 control room was in conformance with the Regulatory Guide 1.97, Revision 3 guidance for those portions examined.

For Unit 1, the inspectors questioned the adequacy of electrical and physical separation of the control room post-accident monitoring components. Redundant channels were physically mounted side by side with electrical wiring from both divisions in a common bundle. This apparent deviation from Section 6 of Supplement 1 to NUREG-0736 requirement regarding Regulatory Guide 1.97 is an unresolved item pending the licensee developing a technical basis for variation from R.G. 1.97 and NRC review of this information. (50-220/88-34-08)

In addition, the Types A, B and C post-accident instrumentation on the control boards was not specifically identified. Regulatory Guide 1.97 Revision 2, section 1.4(B) states, in part, that: "the instruments designated as types A, B and C and categories 1 and 2 should be specifically identified on the control panels...." During this inspection the licensee issued a problem report and operator aid form (88-56) for temporary identification using color dots on accident assessment instrumentation. This apparent deviation from Regulatory Guide 1.97 is an unresolved item pending the licensee developing a technical basis for variations from R.G. 1.97 and NRC review of this information. (50-220/88-34-09)

7.0 Surveillance, Testing and Calibration

Nine Mile Point, Units No. 1 and 2, employ a computerized data base for instrument calibration. Part of the function of this data base is to generate the repetitive maintenance task orders for recalibration of instruments and to provide a historical record of instrument calibrations.

The inspectors reviewed the data base for the instruments inspected, the frequency of calibration, and the date that calibration is next due. Verification was made to determine that procedures are in place for the performance of the calibration. The maintenance and calibration data for Nine Mile Point, Units No. 1 and 2, instrumentation for the inspected variables are found in Attachment 3. The inspectors noted that Unit 1 is in a long term shutdown condition, and the calibration frequency is not a factor in regards to technical specifications. No abnormalities were noted.

8.0 Unresolved Items

Unresolved items are matters for which more information is required in order to ascertain whether they are acceptable, violations, or deviations. Unresolved items are discussed in Sections 4.1, 4.2, 4.5, 5.1, 5.2, 5.3, and 6.0 of this report.

9.0 Exit Meeting

The inspectors met with licensee representatives (denoted in Details, paragraph 1) on November 18, 1988 and discussed the findings for Units No. 1 and 2. For Unit 1 significant items discussed include channel separation, use of isolation devices, tagging and identification of post-accident monitoring instrumentation in the control room, and equipment qualification for the instrumentation inspected.

For Unit 2, significant items include: actual instrument ranges versus documented ranges.

10.0 Follow-Up Technical Meeting

A meeting was held with the licensee to discuss the results of this inspection of R.G. 1.97 activities as it pertains to Unit 1. The meeting was held on December 20, 1988 at NRC/NRR offices in Rockville, Maryland. The licensee, NRR and Region I personnel participated in this meeting.

The primary purpose of this meeting was to discuss the reason for the numerous deviations from R.G. 1.97 noted during this inspection. The licensee explained that it was always their intention to limit their R.G. 1.97 implementation for Unit 1 to specific criteria delineated in Section 6.2 in Supplement 1 to NUREG-0737. They stated that the scope of their R.G. 1.97 implementation was identified in their letter dated

October 18, 1985 from C. V. Mangan of Niagara Mohawk to Domenick B. Vassalo of the NRC. They noted that they had not addressed other criteria identified in R.G. 1.97 (Rev. 2), since they were not explicity called out in Supplement 1 to NUREG-0737. The licensee is currently performing a re-review of Unit 1 to the R.G. 1.97 criteria. Another meeting is scheduled with the licensee on February 3, 1989 to discuss the results of this re-review and licensee plans to address proposed plant modifications associated with some of the R.G. 1.97 deficiencies at Unit 1.

	Description	Paragraph	Docket No. Identification
4.	Drywell atmosphere temperature sensing elements TE201-36A, TE201-50A and TE201-51A have common power supply and no recorder	4.2	50-220/88-34-02
5.	Type A, B & C Post Accident variables not specifically identified on control boards	6.0	50-220/88-34-09
6.	Circuit Isolation Device RI-SC326	4.5	50-220/88-34-07

C. Unit 2 Unresolved Items

1) Vessel level fuel zone (Device No. 2ISC*LT13A,B) FSAR question 421.36-1 indicates range of 230.69 to 430.69. Actual field installed indication is -165 to +35 inches WC. No listing in Table 7.5-1 of FSAR. Paragraph 5.1 (50-410/88-32-01)

2) Service water to diesel generator - Div. I & II (Device No. 2SWP*FI76A&B) FSAR question 421.36-1 indicates a 0-860 gpm range. Drawing C071M indicates a 0-1400 gpm. No listing in FSAR Table 7.5-1. Actual installed field indication 0-1400 x 10^2 gpm.

ESF cooling water flow (Device No. 2SWP*FI13A&B) FSAR question 421.36-1 indicates a 0-10,000 gpm range. FSAR Table 7.5-1 lists a 0-8000 gpm range. Actual installed field indication is 0-8000 gpm.

Service water to diesel generator - Div. III (Device No. 2SWP*FI535) FSAR question 421.36-1 indicates a 0-650 gpm range. Dwg. C071M indicates a 0-1000 gpm range. No listing in FSAR Table 7.5-1. Actual installed field indication 0-1000 x 10^2 gpm. Paragraph 5.2 (50-410/88-32-02)

3) RHR (LPCI) flow (Device No. 2RHS*FI14A,B&C) FSAR question 421.36-1 indicates range of 0-8400 gpm. General Electric dwg. and FSAR Table 7.5-1 indicate a range of 0-10,000 gpm. Actual installed field indicator is 0-8000 gpm. Paragraph 5.3 (50-410/88-32-03)

ATTACHMENT 2

INSTRUMENT SCHEMATICS AND DRAWINGS

I. UNIT NO. 1

A. Reactor Coolant Level

C-22005-C, E21.5 Sheets 1 through 17
C-18015-C, S18.8
C-22004-C, E21.5 Sheets 1 and 8
C-34853-C, E21 Sheet 1
C-34830, Sheet 2
C-34831, Sheet 2
C-23087-C
C-23089-C

B. Reactor Pressure

C-22004-C, E21.5, Sheet 3
C-23077-C, Sheets 1, 2, 3, 6, and 7
C-18015-C, S18.8

C. Drywell Pressure

C-22020-C, E21.5 Sheets 1, 2, and 15
C-18012 Sheet 2
C-18014 Sheet 1
C-22385, 16A and 17A
C-22005-C, E21.5 Sheets 1, 5, 8, and 11

D Drywell Atmosphere Temperature

C-22020-C, E21.5 Sheets 1 and 2
C-18014 Sheet 1 E. Suppression Chamber Water Level (Torus)

E. Suppresion Chamber Water Level (Torus)

C-22020-C, E21.5 Sheet 13
C-22015-C, E21.5 Sheet 14
C-18007-C Sheets 1 and 14

F. Containment Hydrogen/Oxygen Concentration

C-27003-C, E21 Sheet 2
C-27004-C, E21 Sheet 2
C-22020-C, E21.5 Sheets 8 and 11
C-26939-C, S18.9
C-26949-C, S18.9

G. Suppression Pool Water Temperature

C-18014-C, E21 Sheet 1
C-34853-C, E21 Sheets 1 through 6
C-34854-C, E21 Sheets 1 through 6

II. UNIT NO. 2

A. Reactor Vessel Level - Wide Range and Fuel Zone

PID 28B
PID 28C
NSSS 16.020-5003
16.130-001-055
LCR IL2ISC-068
7.241-001-030
7.510-001-183
7.510-001-231
7.510-001-234
7.510-001-249
7.510-001-251
807 E 171TY Sheet 7
EE-3CS
EK-401V
LOOP 2ISC*13
FSK27-19B
FSK27-19C
16.130-001-054
16.130-001-056
LCR IL2ISC-069
7.241-001-011
7.510-001-187
7.510-001-233
7.510-001-240
7.510-001-250
7.241-001-006
EE-11C
EK-401Y
LOOP 2ISC*9

B. Reactor Pressure

PID 28A
LCR IL2ISC-050
7.510-001-250
7.510-001-240
NSSS 16.020-5003F
16.130-001-055
7.241-001-030
807 E 152TY Sheets 14 and 15
7.510-001-251
EE-3CS
7.520-001-447
7.520-001-422
7.242-001-009
EK-401V
FSK 27-19A
LCR IL2ISC-051
7.212-001-057
EE-4R
16.130-001-054
16.130-001-056
7.241-001-006
7.510-001-249
EE-11C
7.212-001-039
7.520-001-454
7.242-001-008
EK-401Y
LOOP 2ISC*6

Containment Atmosphere H_2/O_2 Concentration

PID 82A
FSK 33-02B
LSK 33-02C
LCR IL2CMS-143
LCR IL2CMS-147
7.510-413-281
FSK 33-02A
LSK 33-02A
LCR IL2CMS-142
LCR TL2CMS-146
7.510-413-281C
7.510-413-273

7.510-413-274
7.510-413-276
7.510-413-278
SPEC C001C
LOOP 2CMS-006 Sheets 1 and 2
EE-460CN
DP-384B
LOOP 2CMS*071
7.510-413-275
7.510-413-277
7.510-413-279
CALC 12177-CS-CMS*08
EE-3HJ
EE-460CQ
DP-384AR
CALC 12177-CS-CMS*07

D. Drywell Pressure/ Primary Containment Pressure

12177-CS-CMS*06
FSK 33-02A
LCR IL2CMS-139
LSK 33-02E
EE-3AE
EE-3AL
7.159-401-507
7.159-401-509
7.159-401-126
7.159-401-144
EK-401AB
BK-16HV
7.131-400-004
7.131-400-005
FSK 33D
EE-4J
LOOP 2CMS*1
LOOP 2CMS*2
7.159-401-086
PID 82A
FSK 33-02B
LCR IL2CMS-140
EE-11C
EE-3Q
EE-3RC
7.159-401-508
7.159-401-124
7.159-401-085
EK-401AD
LOOP 2CMS*1
SPEC C071M
7.131-400-003
LCR IL2CMS-008 D.
LCR IL2CMS-009
LSK 33-02D
EK-401Y
7.159-401-145

E. Suppression Pool Water Temperature

PID 82B
FSK 02D
LCR IL2CMS-104
LOOP 2CMS*67
LOOP 2CMS*68
LOOP 2CMS*69
LOOP 2CMS*70
FSK 33-02C
FSK 33-02K
12177-LSK-33-2K
12177-FL2CMS-106
LOOP 2CMS*174
LOOP 2CMS*175

F. Suppression Chamber Pressure

LOOP 2CMS*7

G. Suppression Pool Water Level

LOOP 2CMS*9
LOOP 2CMS*11

H. Drywell Atmosphere Temperature

2 CMS*101
2 CMS*130
2 CMS*116
2 CMS*140

I. Status of Standby Power

12177-TL 2BYS-002
12177-ESK-8BYS13
12177-ESK-7CEC19

J. LPCI/RHR Flow

12177-TL2RHS-009
12177-TL2RHS-010

K. Drywell Spray Flow

12177-TL2RHS-062

L. RHR Heat Exchanger Outlet Temperature

EE114

M. Cooling Water Flow to ESF Components

C071-MAX
C071-MA2
12177-EE-3AE-9
12177-EE-11E-8
12177-TL2SWP-131
12177-EE-3AL-6
12177-EE-11W
807E170TY SH6 and 11
12177-TLRSWP-002
12177-TL2SWP-046
12177-TL2SWP-048
7.159-401-478
7.159-401-488

N. Cooling Water Temperature to ESF System Components

12177-TL2SWP-401
12177-TL2SWP-002
12177-EE-3D-4
12177-EE-3AE-9
12177-EE-1C-5
12177-EE-3AL
12177-EE-11C-5
7.159-401-173
7.159-401-120
7.159-401-506
7.159-401-122
7.159-401-508
213AT/232

ATTACHMENT 3

MAINTENANCE AND CALIBRATION DATA

1. NINE MILE POINT, UNIT NO. 1

	Variable	Frequency	Last Calibration Date	Next Calibration Date
1. Reactor Coolant Level				
	LT 36-03A	18m	3/17/88	Outage 88
	LT 36-03D	18m	3/10/88	Outage 88
	LT 36-33	18m	5/7/88	Outage 90
	LT 36-24A	18m	1/29/88	Outage 88
	LT 36-24B	18m	1/29/88	Outage 88
2. Reactor Pressure				
	PT 36-31	18m	7/12/88	Outage 90
	PT 36-32	18m	7/12/88	Outage 90
3. Drywell Pressure				
	PT 201.2-105	18m	10/27/86	Outage 88
	PT 201.2-106	18m	10/27/86	Outage 88
	PT 201.2-483	Monthly	7/11/88	Outage 88
	PT 201.2-484	Monthly	7/11/88	Outage 88
4. Drywell Atmospheric Temperature				
	TE 201-36A	18m	5/17/86	Outage 88
	TE 201-50A	18m	5/17/86	Outage 88
	TE 201-51A	18m	5/17/86	Outage 88
5. Suppression Chamber Water Level				
	LT 58-04	6m	11/10/88	5/5/98
	LT 58-05	6m	11/10/88	5/5/89
	LT 58-06	6m	11/10/88	5/5/89
6. Containment Hydrogen/Oxygen Concentration				
	201.2-217	Quarterly	1/15/88	Outage 88
	201.2-218	Quarterly	12/2/87	Outage 88
	201.2-330	Quarterly	1/15/88	Outage 88
	201.2-518	Quarterly	12/2/87	Outage 88

7. Suppression Pool Water Temperature

TE 201.2-517 18m	1/29/88	Outage 88
TE 201.2-518 18m	1/29/88	Outage 88

II. NINE MILE POINT, UNIT NO. 2

1. Reactor Coolant Level	Variable	Frequency	Last Calibration Date	Next Calibration Date
	LT 13A	18m	4/14/87	3/1/89
	LT 13B	18m	4/14/87	3/1/89
	LT 9A	18m	5/12/88	3/30/90
	LT 9B	18m	5/12/88	3/30/90
2. Reactor Pressure				
	PT 6A	18m	10/14/88	9/01/90
	PT 6B	18m	10/14/88	9/01/90
3. Drywell Pressure				
	PT 2A	18m	4/6/88	2/21/89
	PT 2B	18m	4/6/88	2/21/89
4. Drywell Atmospheric Temperature				
	TE 116	18m	4/7/87	2/22/89
	TE 117	18m	4/7/87	2/22/89
	TE 118	18m	4/7/87	2/22/89
	TE 119	18m	4/7/87	2/22/89
	TE 120	18m	4/7/87	2/22/89
	TE 121	18m	4/7/87	2/22/89
	TE 122	18m	4/7/87	2/22/89
	TE 123	18m	4/7/87	2/22/89
	TE 124	18m	4/7/87	2/22/89
5. Containment Hydrogen/Oxygen Concentration				
	AIT 71A	Quarterly	8/20/88	12/8/88
	AIT 71B	Quarterly	8/20/88	12/8/88
	AIT 6A	Quarterly	8/26/88	12/13/88
	AIT 6B	Quarterly	8/26/88	12/13/88

6. Suppression Pool Water Level

LT 9A	18m	10/1/88	8/19/90
LT 9B	18m	10/1/88	8/19/90
LT 11A	18m	9/22/88	8/10/90
LT 11B	18m	9/22/88	8/10/90

7. Suppression Pool Water Temperature

TE 67A	18m	9/21/88	8/9/90
TE 68A	18m	9/21/88	8/9/90
TE 69A	18m	9/21/88	8/9/90
TE 70A	18m	9/21/88	8/9/90
TE 67B	18m	9/21/88	8/9/90
TE 68B	18m	9/21/88	8/9/90
TE 69B	18m	9/21/88	8/9/90
TE 70B	18m	9/21/88	8/9/90

8. Primary Containment Pressure

PT 1A	18m	4/7/87	2/22/89
PT 1B	18m	4/7/87	2/22/89

9. Cooling Water Flow to ESF System Components

FT 13A	18m	8/29/87	7/16/89
FT 13B	18m	8/29/87	7/16/89
FT 76A	18m	1/11/88	11/30/89
FT 76B	18m	1/11/88	11/30/89
FT 535	18m	1/16/88	12/3/89

10. Cooling Water Temperature to ESF System Components

TE 31A	18m	9/7/87	7/25/89
TE 31B	18m	9/12/87	7/30/89

11. RHR Heat Exchanger Outlet Temperature

TE 13A	18m	11/2/87	9/19/89
TE 13B	18m	11/2/87	9/19/89

12. LPCI System Flow

FT 14A	18m	4/27/88	3/01/90
FT 14B	18m	4/3/88	2/19/90
FT 14C	18m	4/3/88	2/19/90

13. Drywell Spray Flow

FT 63A	18m	8/24/87	7/11/89
FT 63B	18m	8/24/87	7/11/89

14. Primary Containment Area Radiation

RE 1A	18m	10/19/88	4/22/90
RE 1B	18m	10/19/88	4/22/90
RE 1C	18m	10/19/88	4/22/90
RE 1D	18m	10/19/88	4/22/90

15. Supression Pool Pressure

PT 17A	18m	1/16/88	12/3/89
PT 17B	18m	1/16/88	12/3/89

LP 25

MAY 8 1986

MEMORANDUM FOR: M. Haughey, Project Manager, BWR Project Directorate #3
Division of BWR Licensing

FROM: Gus C. Lainas, Assistant Director for BWR
Division of BWR Licensing

SUBJECT: EMERGENCY RESPONSE CAPABILITY, CONFORMANCE TO R.G. 1.97, REV. 2

Plant Name: Nine Mile Point Nuclear Station, Unit #2
Utility: Niagara Mohawk Power Corporation
Docket Nos.: 50-410
Licensing Status: NTOL
Resp. Directorate: PD #3/DBL
Project Manager: M. Haughey
Review Branch: EICSB/DPA
Review Status: Complete

We are transmitting herewith our subject safety evaluation with its attachment, EG&G Idaho, Inc., Technical Evaluation Report (TER) dated March 1986. We find the instrumentation provided by the Niagara Mohawk Power Corporation for meeting the recommendations of Regulatory Guide 1.97 acceptable, except for the variable neutron flux.

Our safety evaluation is based on our review of EG&G's TER EGG-NTA-7059. This completes EICSB's action on this item.

The EICSB evaluation of licensee performance (SALP input) is also enclosed.

/S/

Gus C. Lainas, Assistant Director for BWR
Division of BWR Licensing

Enclosures:
As stated

cc (w/enc.): E. Adensam, C. E. Rossi, D. Crutchfield, R. W. Houston, J. Mauck, H. Walker, F. Witt, J. Joyce, R. Stevens

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SAFETY EVALUATION
NINE MILE POINT NUCLEAR STATION, UNIT NO. 2
DOCKET NO. 50-410
CONFORMANCE TO REGULATORY GUIDE 1.97

INTRODUCTION AND SUMMARY

Niagara Mohawk Power Corporation (NMPC) was requested by Generic Letter 82-33 to provide a report to the NRC describing how the post-accident monitoring instrumentation meets the guidelines of Regulatory Guide 1.97 as applied to emergency response facilities. The applicant responded to the Regulatory Guide 1.97 portion of the generic letter on October 5, 1984. Additional information was provided by letter dated January 20, 1986 and by Revision 17 of the FSAR.

A detailed review and technical evaluation of the applicant's submittals was performed by EG&G Idaho, Inc., under contract to the NRC, with general supervision by the NRC staff. This work was reported by EG&G in their Technical Evaluation Report (TER), "Conformance to Regulatory Guide 1.97, Nine Mile Point Nuclear Station, Unit No. 2," dated March 1986 (attached). We have reviewed this report and concur with the conclusion that the applicant either conforms to, or is justified in deviating from, the guidance of Regulatory Guide 1.97 for each post-accident monitoring variable, except for the variable neutron flux.

EVALUATION CRITERIA

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983 to answer licensee and applicant questions and concerns regarding the NRC policy on Regulatory Guide 1.97. At these meetings, it was noted that the NRC review would only address exceptions taken to the guidance of Regulatory Guide 1.97. Further, where licensees or

applicants explicitly state that instrument systems conform to the provisions of the regulatory guide, it was noted that no further staff review would be necessary. Therefore, the review performed and reported by EG&G only addresses exceptions to the guidance of Regulatory Guide 1.97. This Safety Evaluation addresses the licensee's submittals based on the review policy described in the NRC regional meetings and the conclusions of the review as reported by EG&G.

EVALUATION

We have reviewed the evaluation performed by our consultant contained in the enclosed TER and concur with its bases and findings. The applicant either conforms to, or has provided an acceptable justification for deviations from the guidance of Regulatory Guide 1.97 for each post-accident monitoring variable except for the variable neutron flux.

Regulatory Guide 1.97 recommends that the neutron flux monitoring instrumentation be Category 1, however existing neutron flux monitoring instrumentation for BWR's is not fully environmentally qualified. A fully qualified Category 1 instrument is presently an industry development item. In his April 1, 1986 letter the applicant committed to monitor industry efforts to develop a qualified Neutron Monitoring System for long-term post-accident monitoring. He further committed that, when qualified equipment becomes available, he would install it at Unit 2 by the first refueling following availability, unless he justifies to the staff that installation of the specific equipment would

result in an overall decrease in the safety of the plant. In this case he would continue his efforts to identify and procure acceptable qualified equipment. The staff finds this commitment acceptable.

CONCLUSION

Based on the staff's review of the enclosed Technical Evaluation Report and the applicant's submittals, we find that the Nine Mile Point Nuclear Station, Unit No. 2, design is acceptable with respect to conformance to Regulatory Guide 1.97, Revision 2, except for the neutron flux instrumentation.

The staff also finds that the existing neutron flux instrumentation is acceptable for interim operation. When qualified neutron flux instrumentation becomes available the applicant should install it at Unit 2 by the first refueling following availability, unless he justifies to the staff that installation of the specific equipment would result in an overall decrease in the safety of the plant.

EICSB SALP INPUT

Plant: Nine Mile Point Nuclear Station, Unit No. 2

Subject: R.G. 1.97/NUREG-0737, Supplement No. 1

Evaluation Criteria	Performance Category	Basis
1. Management	1	Management has been cooperative in establishing communication to clarify their positions. Responses have been timely.
2. Approach to Resolution of Technical Issues	1	The applicant demonstrated an adequate understanding of the issue and provided technically sound justification supporting their positions.
3. Responsiveness	1	The applicant responded to staff request for additional information in a timely manner. Response was adequate.
4. Enforcement History	N/A	No basis for assessment.
5. Reportable Events	N/A	No basis for assessment.
6. Staffing	N/A	No basis for assessment.
7. Training	N/A	No basis for assessment.

ABSTRACT

This EG&G Idaho, Inc., report reviews the submittal for Regulatory Guide 1.97, Revision 3, for Unit No. 2 of the Nine Mile Point Nuclear Station and identifies areas of nonconformance to the regulatory guide. Exceptions to Regulatory Guide 1.97 are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

Docket No. 50-410

CONFORMANCE TO REGULATORY GUIDE 1.97 NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

A. C. Udy

Published March 1986

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6493

CONTENTS

FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of PWR Licensing-A, by EG&G Idaho, Inc., NRR and I&E Support Branch.

The U.S. Nuclear Regulatory Commission funded the work under authorization B&R 20-19-40-41-3.

Docket No. 50-410

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement No. 1, sets forth the documentation to be submitted in a report to the NRC describing how the applicant complies with Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range
2. Environmental qualification
3. Seismic qualification
4. Quality assurance
5. Redundance and sensor location
6. Power supply
7. Location of display
8. Schedule of installation or upgrade

The submittal should identify deviations from the regulatory guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this subject. At these meetings, it was noted that the NRC review would only address exceptions taken to Regulatory Guide 1.97. Where licensees or applicants explicitly state that instrument systems conform to the regulatory guide it was noted that no further staff review would be necessary. Therefore, this

CONFORMANCE TO REGULATORY GUIDE 1.97
NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, the Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement No. 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

The Niagara Mohawk Power Company, the applicant for the Nine Mile Point Nuclear Station, submitted Revision 14 to the Final Safety Analysis Report (FSAR, Reference 4), that included a response to Section 6.2 of the generic letter, on October 5, 1984. Revision 17 of the FSAR updates this information. This information addresses Revision 3 of Regulatory Guide 1.97 (Reference 5). Additional information was provided on January 20, 1986 (Reference 6) and March 3, 1986 (Reference 7).

This report provides an evaluation of this material.

report only addresses exceptions to Regulatory Guide 1.97. The following evaluation is an audit of the applicant's submittals based on the review policy described in the NRC regional meetings.

3. EVALUATION

The applicant provided a response to Item 6.2 of NRC Generic Letter 82-33, on October 5, 1984. The response describes the applicant's position on post-accident monitoring instrumentation in answering FSAR question F421.36(7.5). This evaluation is based on Revision 17 of the FSAR and on additional information provided on January 20, 1986 and March 3, 1986.

3.1 Adherence to Regulatory Guide 1.97

The applicant has provided a review of their post-accident monitoring instrumentation that compares the instrumentation characteristics against the recommendations of Regulatory Guide 1.97, Revision 3 (Reference 5). Table 1.8-1 of the FSAR states that Unit No. 2 complies with the regulatory intent of Regulatory Guide 1.97. We understand that the instrumentation presently identified for Regulatory Guide 1.97 will be operational prior to the units' initial fuel load. Therefore, we conclude that the applicant has provided an explicit commitment on conformance to Regulatory Guide 1.97. Exceptions to and deviations from the regulatory guide are noted in Section 3.3.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide the information required to permit the control room operator to take specific manually controlled safety actions. The applicant has identified the following Type A variables.

1. Containment hydrogen concentration
2. Containment oxygen concentration
3. Reactor vessel pressure
4. Reactor vessel level

5. Suppression pool water temperature

6. Drywell atmosphere temperature

7. Drywell atmosphere pressure

The above variables meet the Category 1 recommendations as required for Type A variables except as listed in Section 3.3.

3.3 Exceptions to Regulatory Guide 1.97

The applicant identified deviations and exceptions from Regulatory Guide 1.97. These are discussed in the following paragraphs.

3.3.1 Neutron Flux

Regulatory Guide 1.97 recommends Category 1 instrumentation with a range of 10^{-6} to 100 percent of full power for this variable. There are three sets of instrumentation with overlapping ranges. The licensee states that the instrumentation is seismically qualified and environmentally qualified for short term post-accident operation. The power range has a range of 0.5 to 125 percent; the intermediate range has a range of 4.0×10^{-5} to 12.6 percent (1×10^{8} to 1.5×10^{13} nu); the source range has a range of 1×10^{3} to 5×10^{9} nu.

The power sources for the instrumentation is as follows:

Source and intermediate range monitors--24/48 Vdc (battery chargers and batteries)

Linear and average power range monitors--110 Vac from reactor protection system uninterruptable power supplies with normal, alternate and battery sources.

In the process of our review of the neutron flux instrumentation for boiling water reactors, we note that the detectors and their cables are not fully environmentally qualified as required by Regulatory Guide 1.97. A Category 1 system that meets all the criteria of Regulatory Guide 1.97 in an industry development item. Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The applicant has commited to follow industry development of this equipment, and provide further information for this variable prior to the completion of the first refueling outage.

3.3.2 Reactor Coolant System Soluble Boron Concentration

Regulatory Guide 1.97 recommends a range of 0 to 1000 parts per million for this variable. Reference 6 identifies the range as 50 to 2000 parts per million.

The applicant deviates from Regulatory Guide 1.97 with respect to the range of this post-accident sampling capability. This deviation goes beyond the scope of this review and is being addressed by the NRC as part of their review of NUREG-0737, Item, II.B.3.

3.3.3 Coolant Level in Reactor

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the bottom of the core support plate to the centerline of the main steamline. The applicant identifies fuel zone instrumentation with a range of 230.69 to 430.69 inches, and wide range instrumentation with a range of 375.7 to 587.7 inches (above vessel zero). Thus, the range covered is 62.3 inches short of reaching the centerline of the main steamline. All safety trips occur within these two ranges.

With all safety trips occurring before the range would be exceeded, there are no additional manual operations to be taken should the level range of the above instrumentation be exceeded.

3.3.6 Primary Containment Isolation Valve Position

Regulatory Guide 1.97 recommends Category 1 indication for this variable. With the exception of the transversing incore probe system isolation valves, the instrumentation identified by the applicant for this variable meets the Category 1 requirements. Check valves are specifically excluded by Regulatory Guide 1.97.

The transversing incore probe (TIP) system isolation valves consist of ball valves (used when probe is out of the guide tube) and shear valves (detonated if the probe is in the guide tube). The TIPs are normally withdrawn with the ball valves closed. The shear valves are operated manually if a probe is in a guide tube. As these valves are non-safety related, we find this exception from Regulatory Guide 1.97 acceptable.

3.3.7 Radiation Level in Circulating Primary Coolant

The applicant states that radiation level measurements to indicate fuel cladding failure are provided by the following instruments:

1. Post-accident sampling system
2. Condenser off-gas radiation monitors
3. Main steamline radiation monitors
4. Primary containment radiation monitors
5. Containment hydrogen concentration monitors

Based on the alternate instrumentation provided by the applicant, we conclude that the instrumentation supplied for this variable is adequate and, therefore, acceptable.

In addition, there is a Category 3 instrument covering from 525.7 to 925.7 inches. Besides this channel, there is a Category 3 recorder that monitors from 525.7 to 705.7 inches.

Based on the applicant's justification and Category 3 instrumentation that monitors levels beyond the fuel zone and the wide range instruments, we find the instrumentation provided for this variable acceptable.

3.3.4 Drywell Pressure

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. Thus, redundant Class 1E power sources should be provided. Reference 4 identifies Division 1 Class 1E power for all this instrumentation. Reference 6 corrects this, stating that divisional independence is maintained by separate Class 1E power sources. We find this acceptable.

3.3.5 Drywell Sump Level
Drywell Drain Sumps Level

Regulatory Guide 1.97 recommends Category 1 instrumentation for these variables with a range from the bottom to the top. The applicant states, in Reference 6, that Category 3 instruments are used for this variable.

We conclude that the instrumentation provided by the licensee will provide appropriate monitoring of the parameters of concern. This is based on (a) for small leaks, the instrumentation is not expected to experience harsh environments during operation, (b) for larger leaks, the sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, thus negating the drywell sump level and drywell drain sumps level instrumentation and (c) this instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety-related system in a post-accident situation. Therefore, we find the Category 3 instrumentation provided acceptable.

3.3.8 Analysis of Primary Coolant

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 10^{-6} to 10 Ci/gm. The applicant, in Reference 6, identifies the range as 10^{-6} to 10 Ci/gm. This range satisfies the regulatory guide recommendation.

3.3.9 Radiation Exposure Rate

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 10^{-1} to 10^{4} R/hr. The licensee has instrumentation that has a range of either 10^{-1} to 10^{4} mR/hr (10^{-4} to 10 R/hr) or 10^{-2} to 10^{3} mR/hr (10^{-5} to 1 R/hr). A deviation exists in the upper limit of the range. The applicant states that when accessibility is re-established to service safety-related equipment, it is done by post-accident sampling and portable instrumentation.

From a radiological standpoint, if the radiation levels reach or exceed the upper limit of the range, personnel would not be permitted into the areas without portable monitoring (except for life saving). Based on the alternate instrumentation used by the applicant with this variable, we find the proposed ranges for the radiation exposure rate monitors acceptable.

3.3.10 Suppression Pool Water Temperature

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 40 to 230°F. The instrumentation supplied by the applicant has a range of 50 to 250°F. The deviation of the lower limit of the range is 10° out of 250° or 4 percent. Considering instrument accuracy, we find this deviation minor and acceptable.

3.3.11 Drywell Atmosphere Temperature

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 40 to 440°F. The instrumentation provided by the licensee for this variable has a range of 0 to 400°F.

Section 6.2.1.1.1 of the FSAR identifies the maximum drywell atmosphere temperature as 340°F. Since the worst case postulated accident will not increase the drywell atmosphere temperature above 340°F, we find the range of 0 to 400°F adequate to monitor this variable during all accident and post-accident conditions.

3.3.12 Residual Heat Removal Heat Exchanger Outlet Temperature

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. In Reference 4, the licensee instrumentation was not identified as environmentally qualified. Reference 6 identifies environmentally qualified instrumentation that is Category 2. We find this acceptable.

3.3.13 Cooling Water Temperature to Engineered Safety Feature System Components

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 32 to 200°F. The applicant's instrumentation has a range of 35 to 130°F. The licensee states that this range is adequate because the service water is always between 38 and 77°F, the temperature limits of the water from Lake Ontario.

We find this temperature range acceptable for this once through system.

3.3.14 Secondary Containment Area Radiation

The applicant states that this variable need not be implemented. The applicant reports that the use of local radiation monitors to detect breach or leakage through primary containment penetrations results in ambiguous indications. This is due to the radioactivity in the primary containment,

the radioactivity in the fluids flowing in emergency core coolant system piping and the amount and location of fluid and electrical penetrations. The applicant concludes that the use of the reactor building exhaust gaseous and the stack gaseous effluent monitors is the proper way to accomplish the detection of releases, release assessment and long term surveillance recommended for this variable. The applicant states that these gas effluent monitors are adequate to monitor this variable. We find the alternate instrumentation provided is acceptable for this variable.

3.3.15 Noble Gas-Radwaste Vent Noble Gas-Common Plant Vent

Regulatory Guide 1.97 recommends instrumentation for these variables with ranges of 10^{-6} to 10^{3} μCi/cc for the radwaste vent and 10^{-6} to 10^{4} μCi/cc for the common plant vent. In Reference 6, the applicant states the range of both of these variables is in conformance with the regulatory guide.

3.3.16 Plant and Envrions Radiation

Regulatory Guide 1.97 recommends portable instrumentation for this variable with ranges of 10^{-3} to 10^{4} R/hr-photons, and 10^{-3} to 10^{4} rads/hr-beta and low energy photons. The applicant's instrumentation covers ranges of 10^{-3} to 50 R/hr low energy gamma and beta and 10^{-2} to 10^{3} R/hr gamma. The licensee states that this combination is adequate to monitor the dose rates expected post-accident.

As the applicant has determined that the instrument range is adequate, the supplied instrumentation is acceptable.

3.3.17 Accident Sampling (Primary Coolant, Containment Air and Sump)

The applicant's post-accident sampling facility provides sampling and analysis. However, the range for dissolved gas analysis does not meet the recommended range of 0 to 2000 cc/kg.

The applicant deviates from Regulatory Guide 1.97 with respect to the range of this post-accident sampling capability. This deviation goes beyond the scope of this review and has been addressed by the NRC as part of their review of NUREG-0737, Item II.B.3.

4. CONCLUSIONS

Based on our review, we find that the applicant either conforms to or is justified in deviating from Regulatory Guide 1.97, with the following exception:

1. Neutron flux--the applicant's present instrumentation is acceptable on an interim basis until Category 1 instrumentation is developed and installed. The applicant should commit to install Category 1 instrumentation for this variable when it becomes available (Section 3.3.1).

5. REFERENCES

1. NRC letter, D. G. Eisenhut to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, NRC, Office of Standards Development, December 1980.

3. Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737, Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Niagara Mohawk Power Corporation letter, C. V. Mangan to D. G. Eisenhut, NRC, "Amendment 14 to Application to Operating Licensee," October 5, 1984, NMP2L 0169.

5. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 3, NRC, Office of Nuclear Regulatory Research, May 1983.

6. Niagara Mohawk Power Corporation letter, C. V. Mangan to E. G. Adensam, NRC, January 20, 1986, NMP2L 0589.

7. Niagara Mohawk Power Corporation letter, C. V. Mangan to E. G. Adensam, NRC, March 3, 1986, NMP2L 0644.

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12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

This EG&G Idaho, Inc. report reviews the submittals for Unit No. 2 of the Nine Mile Point Nuclear Station, and identifies areas of nonconformance to Regulatory Guide 1.97. Exceptions to these guidelines are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

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Safety Evaluation Report

related to the operation of
Nine Mile Point Nuclear Station,
Unit No. 2

Docket No. 50-410

Niagara Mohawk Power Corporation
Rochester Gas and Electric Corporation
Central Hudson Gas and Electric Corporation
New York State Electric and Gas Corporation
Long Island Lighting Company

U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

September 1986

The digital isolator utilizes a Hewlett-Packard HCPL-2630 optocoupler located in the safety isolation modules (SIMs). The HCPL-2630 is the Class 1E to non-Class 1E boundary, and provides the required electrical isolation between safety-related and non-safety-related systems. On application of the MCF, the communication link between the SIM and the ICU changed by two least-significant digits; after 1 sec, the indication returned to its pretest reading. The SIM was damaged by the test.

The analog isolator utilizes an Intronic Model 1A-184 isolation amplifier module to provide the required electrical isolation between safety-related and non-safety-related systems. This amplifier is located in the analog isolation modules (AIMs). On the application of the MCF (as stated above), the communication link between the AIMs and the ICU changed by one least-significant digit. This change was permanent. The staff found the significant-digit changes for both the digital and analog isolators acceptable.

However, further evaluation of the April 15, 1986, submittal showed that the Kaman analog isolation device is a three-output device and that the applicant had only tested the output being used and not the other two. The untested outputs were of concern to the staff. As a result, the staff requested that the applicant test these outputs or render the connections inaccessible to the MCF. By letter dated May 9, 1986, the applicant stated that the unused output connections would be rendered inaccessible to the MCF and further agreed to test any spare output connection should its use ever become necessary. The applicant further agreed to notify the staff of the test results before the two untested output connections are used.

The tests performed by the applicant demonstrated the ability of the Kaman isolation modules to effectively stop the propagation of the MCF from output to input and to adequately protect the Class 1E inputs. Therefore, the staff concludes that these devices may be used as electrical isolation devices between Class 1E signals and non-safety-related systems and between redundant safety systems. This resolves Confirmatory Issue 20 concerning the isolation of circuits.

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings

7.5.2.2 Postaccident Monitoring Instrumentation

Generic Letter 82-33 requested that applicants provide a report to the NRC staff describing how the postaccident monitoring instrumentation meets RG 1.97 as it applies to emergency response facilities. The applicant responded to the RG 1.97 portion of the generic letter on October 5, 1984, and supplied additional information by letter dated January 20, 1986, and by FSAR Amendment 17.

EG&G Idaho, Inc., under contract to the NRC with general supervision by the NRC staff, performed a detailed review and technical evaluation of the applicant's submittals. This work was reported by EG&G in its Technical Evaluation Report (TER), "Conformance to Regulatory Guide 1.97, Nine Mile Point Nuclear Station, Unit 2," dated March 1986 (attached as Appendix M to this supplement). The

staff has reviewed this report and concurs with the conclusion that the applicant either conforms to, or is justified in deviating from, RG 1.97 for each postaccident monitoring variable except neutron flux.

After the generic letter was issued, the NRC staff held regional meetings in February and March 1983 to answer licensee and applicant questions and concerns regarding the NRC policy on RG 1.97. At these meetings, it was noted that the NRC review would only address exceptions taken to RG 1.97. Furthermore, where licensees or applicants explicitly state that instrument systems conform to the regulatory guide, no further staff review would be necessary. Therefore, the review performed and reported by EG&G only addresses exceptions to RG 1.97. This safety evaluation addresses the applicant's submittals on the basis of the review policy described in the NRC regional meetings and the conclusions of the review as reported by EG&G.

The staff has reviewed the evaluation performed by its consultant as discussed in the TER attached as Appendix M to this supplement. The staff concurs with its bases and findings. The applicant either conforms to, or has provided an acceptable justification for deviations from, RG 1.97 for each postaccident monitoring variable except neutron flux.

RG 1.97 recommends that the neutron flux monitoring instrumentation be Category 1; however, existing neutron flux monitoring instrumentation for boiling-water reactors is not fully environmentally qualified. A fully qualified Category 1 instrument currently is an industry development item. By letter dated April 1, 1986, the applicant committed to monitor industry efforts to develop a qualified neutron monitoring system for long-term postaccident monitoring. The applicant further committed to install qualified equipment at NMP-2 by the first refueling outage after the qualified equipment became available, unless the applicant justifies to the staff that installation of the specific equipment would result in an overall decrease in the safety of the plant. In this case the applicant would continue efforts to identify and procure acceptable qualified equipment. The staff finds this commitment acceptable.

On the basis of its review of the TER (Appendix M) and the applicant's submittals, the staff finds that the NMP-2 design is acceptable with respect to conformance to RG 1.97, Revision 2, except for the neutron flux instrumentation.

The staff also finds that the existing neutron flux instrumentation is acceptable for interim operation. When qualified neutron flux instrumentation becomes available, the applicant should install it at Unit 2 by the first refueling following availability, unless the applicant justifies to the staff that installation of the specific equipment would result in an overall decrease in the safety of the plant.

APPENDIX M

CONFORMANCE TO REGULATORY GUIDE 1.97, NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

CONFORMANCE TO REGULATORY GUIDE 1.97
NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

A. C. Udy

Published March 1986

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-761D01570
FIN No. A6493

ABSTRACT

This EG&G Idaho, Inc., report reviews the submittal for Regulatory Guide 1.97, Revision 3, for Unit No. 2 of the Nine Mile Point Nuclear Station and identifies areas of nonconformance to the regulatory guide. Exceptions to Regulatory Guide 1.97 are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

Docket No. 50-410

FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of PWR Licensing-A, by EG&G Idaho, Inc., NRR and I&E Support Branch.

The U.S. Nuclear Regulatory Commission funded the work under authorization B&R 20-19-40-41-3.

Docket No. 50-410

CONTENTS

CONFORMANCE TO REGULATORY GUIDE 1.97
NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, the Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement No. 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

The Niagara Mohawk Power Company, the applicant for the Nine Mile Point Nuclear Station, submitted Revision 14 to the Final Safety Analysis Report (FSAR, Reference 4), that included a response to Section 6.2 of the generic letter, on October 5, 1984. Revision 17 of the FSAR updates this information. This information addresses Revision 3 of Regulatory Guide 1.97 (Reference 5). Additional information was provided on January 20, 1986 (Reference 6) and March 3, 1986 (Reference 7).

This report provides an evaluation of this material.

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement No. 1, sets forth the documentation to be submitted in a report to the NRC describing how the applicant complies with Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range

2. Environmental qualification

3. Seismic qualification

4. Quality assurance

5. Redundance and sensor location

6. Power supply

7. Location of display

8. Schedule of installation or upgrade

The submittal should identify deviations from the regulatory guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this subject. At these meetings, it was noted that the NRC review would only address exceptions taken to Regulatory Guide 1.97. Where licensees or applicants explicitly state that instrument systems conform to the regulatory guide it was noted that no further staff review would be necessary. Therefore, this

report only addresses exceptions to Regulatory Guide 1.97. The following evaluation is an audit of the applicant's submittals based on the review policy described in the NRC regional meetings.

3. EVALUATION

The applicant provided a response to Item 6.2 of NRC Generic Letter 82-33, on October 5, 1984. The response describes the applicant's position on post-accident monitoring instrumentation in answering FSAR question F421.36(7.5). This evaluation is based on Revision 17 of the FSAR and on additional information provided on January 20, 1986 and March 3, 1986.

3.1 Adherence to Regulatory Guide 1.97

The applicant has provided a review of their post-accident monitoring instrumentation that compares the instrumentation characteristics against the recommendations of Regulatory Guide 1.97, Revision 3 (Reference 5). Table 1.8-1 of the FSAR states that Unit No. 2 complies with the regulatory intent of Regulatory Guide 1.97. We understand that the instrumentation presently identified for Regulatory Guide 1.97 will be operational prior to the units' initial fuel load. Therefore, we conclude that the applicant has provided an explicit commitment on conformance to Regulatory Guide 1.97. Exceptions to and deviations from the regulatory guide are noted in Section 3.3.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide the information required to permit the control room operator to take specific manually controlled safety actions. The applicant has identified the following Type A variables.

1. Containment hydrogen concentration

2. Containment oxygen concentration

3. Reactor vessel pressure

4. Reactor vessel level

5. Suppression pool water temperature

6. Drywell atmosphere temperature

7. Drywell atmosphere pressure

The above variables meet the Category 1 recommendations as required for Type A variables except as listed in Section 3.3.

3.3 Exceptions to Regulatory Guide 1.97

The applicant identified deviations and exceptions from Regulatory Guide 1.97. These are discussed in the following paragraphs.

3.3.1 Neutron Flux

Regulatory Guide 1.97 recommends Category 1 instrumentation with a range of 10^{-6} to 100 percent of full power for this variable. There are three sets of instrumentation with overlapping ranges. The licensee states that the instrumentation is seismically qualified and environmentally qualified for short term post-accident operation. The power range has a range of 0.5 to 125 percent; the intermediate range has a range of 4.0×10^{-5} to 12.6 percent (1×10^{8} to 1.5×10^{13} nu); the source range has a range of 1×10^{3} to 5×10^{9} nu.

The power sources for the instrumentation is as follows:

Source and intermediate range monitors--24/48 Vdc (battery chargers and batteries)

Linear and average power range monitors--110 Vac from reactor protection system uninterruptable power supplies with normal, alternate and battery sources.

In the process of our review of the neutron flux instrumentation for boiling water reactors, we note that the detectors and their cables are not fully environmentally qualified as required by Regulatory Guide 1.97. A Category 1 system that meets all the criteria of Regulatory Guide 1.97 in an industry development item. Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The applicant has commited to follow industry development of this equipment, and provide further information for this variable prior to the completion of the first refueling outage.

3.3.2 Reactor Coolant System Soluble Boron Concentration

Regulatory Guide 1.97 recommends a range of 0 to 1000 parts per million for this variable. Reference 6 identifies the range as 50 to 2000 parts per million.

The applicant deviates from Regulatory Guide 1.97 with respect to the range of this post-accident sampling capability. This deviation goes beyond the scope of this review and is being addressed by the NRC as part of their review of NUREG-0737, Item, II.B.3.

3.3.3 Coolant Level in Reactor

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the bottom of the core support plate to the centerline of the main steamline. The applicant identifies fuel zone instrumentation with a range of 230.69 to 430.69 inches, and wide range instrumentation with a range of 375.7 to 587.7 inches (above vessel zero). Thus, the range covered is 62.3 inches short of reaching the centerline of the main steamline. All safety trips occur within these two ranges.

With all safety trips occurring before the range would be exceeded, there are no additional manual operations to be taken should the level range of the above instrumentation be exceeded.

In addition, there is a Category 3 instrument covering from 525.7 to 925.7 inches. Besides this channel, there is a Category 3 recorder that monitors from 525.7 to 705.7 inches.

Based on the applicant's justification and Category 3 instrumentation that monitors levels beyond the fuel zone and the wide range instruments, we find the instrumentation provided for this variable acceptable.

3.3.4 Drywell Pressure

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. Thus, redundant Class 1E power sources should be provided. Reference 4 identifies Division 1 Class 1E power for all this instrumentation. Reference 6 corrects this, stating that divisional independence is maintained by separate Class 1E power sources. We find this acceptable.

3.3.5 Drywell Sump Level
Drywell Drain Sumps Level

Regulatory Guide 1.97 recommends Category 1 instrumentation for these variables with a range from the bottom to the top. The applicant states, in Reference 6, that Category 3 instruments are used for this variable.

We conclude that the instrumentation provided by the licensee will provide appropriate monitoring of the parameters of concern. This is based on (a) for small leaks, the instrumentation is not expected to experience harsh environments during operation, (b) for larger leaks, the sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, thus negating the drywell sump level and drywell drain sumps level instrumentation and (c) this instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety-related system in a post-accident situation. Therefore, we find the Category 3 instrumentation provided acceptable.

3.3.6 Primary Containment Isolation Valve Position

Regulatory Guide 1.97 recommends Category 1 indication for this variable. With the exception of the transversing incore probe system isolation valves, the instrumentation identified by the applicant for this variable meets the Category 1 requirements. Check valves are specifically excluded by Regulatory Guide 1.97.

The transversing incore probe (TIP) system isolation valves consist of ball valves (used when probe is out of the guide tube) and shear valves (detonated if the probe is in the guide tube). The TIPs are normally withdrawn with the ball valves closed. The shear valves are operated manually if a probe is in a guide tube. As these valves are non-safety related, we find this exception from Regulatory Guide 1.97 acceptable.

3.3.7 Radiation Level in Circulating Primary Coolant

The applicant states that radiation level measurements to indicate fuel cladding failure are provided by the following instruments:

1. Post-accident sampling system

2. Condenser off-gas radiation monitors

3. Main steamline radiation monitors

4. Primary containment radiation monitors

5. Containment hydrogen concentration monitors

Based on the alternate instrumentation provided by the applicant, we conclude that the instrumentation supplied for this variable is adequate and, therefore, acceptable.

3.3.8 Analysis of Primary Coolant

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 10^{-6} to 10 Ci/gm. The applicant, in Reference 6, identifies the range as 10^{-6} to 10 Ci/gm. This range satisfies the regulatory guide recommendation.

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Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 10^{-1} to 10^{4} R/hr. The licensee has instrumentation that has a range of either 10^{-1} to 10^{4} mR/hr (10^{-4} to 10 R/hr) or 10^{-2} to 10^{3} mR/hr (10^{-5} to 1 R/hr). A deviation exists in the upper limit of the range. The applicant states that when accessibility is re-established to service safety-related equipment, it is done by post-accident sampling and portable instrumentation.

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Section 6.2.1.1.1 of the FSAR identifies the maximum drywell atmosphere temperature as 340°F. Since the worst case postulated accident will not increase the drywell atmosphere temperature above 340°F, we find the range of 0 to 400°F adequate to monitor this variable during all accident and post-accident conditions.

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Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. In Reference 4, the licensee instrumentation was not identified as environmentally qualified. Reference 6 identifies environmentally qualified instrumentation that is Category 2. We find this acceptable.

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Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 32 to 200°F. The applicant's instrumentation has a range of 35 to 130°F. The licensee states that this range is adequate because the service water is always between 38 and 77°F, the temperature limits of the water from Lake Ontario.

We find this temperature range acceptable for this once through system.

3.3.14 Secondary Containment Area Radiation

The applicant states that this variable need not be implemented. The applicant reports that the use of local radiation monitors to detect breach or leakage through primary containment penetrations results in ambiguous indications. This is due to the radioactivity in the primary containment,

the radioactivity in the fluids flowing in emergency core coolant system piping and the amount and location of fluid and electrical penetrations. The applicant concludes that the use of the reactor building exhaust gaseous and the stack gaseous effluent monitors is the proper way to accomplish the detection of releases, release assessment and long term surveillance recommended for this variable. The applicant states that these gas effluent monitors are adequate to monitor this variable. We find the alternate instrumentation provided is acceptable for this variable.

3.3.15 Noble Gas-Radwaste Vent Noble Gas-Common Plant Vent

Regulatory Guide 1.97 recommends instrumentation for these variables with ranges of 10^{-6} to 10^{3} μCi/cc for the radwaste vent and 10^{-6} to 10^{4} μCi/cc for the common plant vent. In Reference 6, the applicant states the range of both of these variables is in conformance with the regulatory guide.

3.3.16 Plant and Envrions Radiation

Regulatory Guide 1.97 recommends portable instrumentation for this variable with ranges of 10^{-3} to 10^{4} R/hr-photons, and 10^{-3} to 10^{4} rads/hr-beta and low energy photons. The applicant's instrumentation covers ranges of 10^{-3} to 50 R/hr low energy gamma and beta and 10^{-2} to 10^{3} R/hr gamma. The licensee states that this combination is adequate to monitor the dose rates expected post-accident.

As the applicant has determined that the instrument range is adequate, the supplied instrumentation is acceptable.

3.3.17 Accident Sampling (Primary Coolant, Containment Air and Sump)

The applicant's post-accident sampling facility provides sampling and analysis. However, the range for dissolved gas analysis does not meet the recommended range of 0 to 2000 cc/kg.

The applicant deviates from Regulatory Guide 1.97 with respect to the range of this post-accident sampling capability. This deviation goes beyond the scope of this review and has been addressed by the NRC as part of their review of NUREG-0737, Item II.B.3.

4. CONCLUSIONS

Based on our review, we find that the applicant either conforms to or is justified in deviating from Regulatory Guide 1.97, with the following exception:

1. Neutron flux--the applicant's present instrumentation is acceptable on an interim basis until Category 1 instrumentation is developed and installed. The applicant should commit to install Category 1 instrumentation for this variable when it becomes available (Section 3.3.1).

5. REFERENCES

1. NRC letter, D. G. Eisenhut to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, NRC, Office of Standards Development, December 1980.

3. Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737, Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Niagara Mohawk Power Corporation letter, C. V. Mangan to D. G. Eisenhut, NRC, "Amendment 14 to Application to Operating Licensee," October 5, 1984, NMP2L 0169.

5. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 3, NRC, Office of Nuclear Regulatory Research, May 1983.

6. Niagara Mohawk Power Corporation letter, C. V. Mangan to E. G. Adensam, NRC, January 20, 1986, NMP2L 0589.

7. Niagara Mohawk Power Corporation letter, C. V. Mangan to E. G. Adensam, NRC, March 3, 1986, NMP2L 0644.

NRC FORM 335 (2-84) NRCM 1102 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

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Conformance to Regulatory Guide 1.97,
Nine Mile Point Nuclear Station,
Unit No. 2

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4 DATE REPORT COMPLETED

MONTH: March | YEAR: 1986

5 AUTHOR(S)

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EG&G Idaho, Inc.
Idaho Falls, ID 83415

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Division of PWR Licensing - A
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a TYPE OF REPORT

Technical Evaluation Report

b PERIOD COVERED (Inclusive dates)

12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

This EG&G Idaho, Inc. report reviews the submittals for Unit No. 2 of the Nine Mile Point Nuclear Station, and identifies areas of nonconformance to Regulatory Guide 1.97. Exceptions to these guidelines are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

14 DOCUMENT ANALYSIS - a KEYWORDS DESCRIPTORS

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MAR 18 1986

MEMORANDUM FOR: Elinor G. Adensam, Director, BWR Project Directorate #3
Division of BWR Licensing

FROM: F. Rosa, Chief
Electrical, Instrumentation & Control Systems Branch
Division of PWR Licensing-A

SUBJECT: INSTALLATION OF A CATEGORY 1 (IN ACCORDANCE WITH R.G. 1.97) NEUTRON FLUX INSTRUMENTATION AT NINE MILE POINT UNIT 2

As requested by the Nine Mile Point Unit 2 Project Manager (M. Haughey) the purpose of this memorandum is to document our position on the installation of the subject instrumentation at NMP-2. Because this may be an appeal item Ms. Haughey requested this memorandum ahead of our formal SER so that she can pursue this item with the applicant in a timely manner prior to licensing.

R.G. 1.97 identifies neutron flux as one of the variables required to be monitored during and following an accident. The design and qualification criteria for the associated instrumentation is listed as Category 1 in the guide. One of the qualification criteria for a Category 1 instrument as defined by R.G. 1.97, is that it be environmentally qualified. The applicant indicated in his January 20, 1986 letter that the environmental qualification of his neutron flux instrumentation under harsh environment conditions is for a limited time. In subsequent conversations with him he indicated that the amount of time the instrumentation could survive in a harsh environment is only a matter of seconds. It is our position that, in accordance with R.G. 1.97, the environmental qualification of this instrumentation should ensure its survivability during and following an accident.

Because a Category 1 neutron flux instrument is presently not available but is an industry development item, we have been requiring utilities to follow the development of this hardware and install a Category 1 instrument when it becomes available. We have presently not received a commitment from the NMP-2 applicant to install a Category 1 instrument when it becomes available. We therefore intend to condition the NMP-2 license that this be done. Our position has been discussed with the NMP-2 applicant.

Original Signed By
Faust Rosa*

Faust Rosa, Chief
Electrical, Instrumentation &
Control Systems Branch
Division of PWR Licensing-A

cc: See attached list

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

April 1, 1986
(NMP2L 0676)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

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

During discussions regarding Regulatory Guide 1.97, your staff requested that we identify our plans for installing a Neutron Monitoring System at Nine Mile Point Unit 2. This would be qualified to withstand a post-accident environment.

In principle, Niagara Mohawk believes such a system would be beneficial and is predisposed to install an environmentally qualified system should a thorough evaluation demonstrate the system's qualification, performance and safety. At the present time, however, we are not aware of any system available that has been demonstrated to meet that criteria, and, therefore, we have no current plans to install such a system at Unit 2.

Niagara Mohawk will continue to monitor industry efforts to develop a qualified Neutron Monitoring System for long-term post-accident monitoring. When qualified equipment becomes available, Niagara Mohawk will install it at Unit 2 by the first refueling following availability, unless we justify to the Nuclear Regulatory Commission that installation of the specific equipment would result in an overall decrease in the safety of the plant. In this case, Niagara Mohawk will continue its efforts to identify and procure acceptable qualified equipment.

Very truly yours,

C. V. Mangan
Senior Vice President

AFZ:ar
1418G

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

March 3, 1986
(NMP2L 0644)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

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

Attached is additional information concerning the Nine Mile Point Unit 2 compliance to Regulatory Guide 1.97. This response is in addition to the response provided on January 20, 1986.

This material was discussed in telephone conversations with your staff on February 13, 1986 and February 21, 1986. Niagara Mohawk expects that this material provides the necessary information to close Confirmatory Item 10.

Very truly yours,

C. V. Mangan
Senior Vice President

TL:ja
1361G
Attachment

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

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3 pp.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)

Niagara Mohawk Power Corporation) Docket No. 50-410

(Nine Mile Point Unit 2))

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Senior Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

C. V. Mangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Oneonda, this 3rd day of March, 1986.

Notary Public in and for
Oneonda County, New York

My Commission expires:

Notary P...
Qualified in ...
My ...

87

2. Neutron flux - The applicant should provide Class 1E power sources for this instrumentation; the applicant should show that the source and intermediate ranges have sufficient overlap.

Response

In addition to the information provided in our January 20, 1986 letter, Niagara Mohawk will continue to follow and evaluate developments in the nuclear industry concerning neutron flux monitoring instrumentation. Prior to the conclusion of the first refueling outage, Niagara Mohawk will update and identify to the Nuclear Regulatory Commission the status of the neutron flux monitoring instrumentation issue for Nine Mile Point Unit 2.

4. Coolant level in reactor - The applicant should identify the remainder of this instrumentation in accordance with Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations identified.

Response

As stated in our January 20, 1986 letter, conformance to Regulatory Guide 1.97, Revision 3 is accomplished by the use of two transmitters per division, one fuel zone and one wide range. As noted in Section 3.3.3 of your November 15, 1985 letter, the wide range transmitters were omitted from Table 421.36-1 and will be added in the next Final Safety Analysis Report amendment.

The wide range transmitters (2ISC*LT9C and D) are calibrated to monitor the 375.70 in. level, which is inside the fuel zone transmitter range, to the 585.70 in. level which is 62.3 in. below the centerline of the main steam lines at 648 in. It should be noted that all safety trips from reactor level occur within these level ranges.

This range meets the intent of the regulatory guide which is to restore and maintain reactor pressure vessel water level to ensure adequate core cooling.

Water level indication is available in the control room to the operator from one transmitter (2ISC*LT105) covering the 525.70 to 925.70 in. level which is well above the main steam lines. A second transmitter (2ISC*PDT110) covering the 525.70 to 705.70 in. level is also available to the control room operator on a strip chart recorder. However, neither of these transmitters (2ISC*PDT110 and LT105) fully meet the qualification requirements of Regulatory Guide 1.97.

LP-25

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION / 300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202 / TELEPHONE (315) 474-1511

January 20, 1986
(NMP2L 0589)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

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

Attached is the Nine Mile Point Unit 2 response to the letter from W. Butler (NRC) to B. G. Hooten (NMPC), dated November 15, 1986 concerning conformance to Regulatory Guide 1.97.

This material provides the information necessary to close Confirmatory Item 10.

Very truly yours,

C. V. Mangan
Senior Vice President

TRL:ja
Attachment

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

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8 pp

ATTACHMENT

1. The licensee should identify plant-specific Type A variables and verify that the instrumentation for them is Category I.

   Response

   Table 421.36-1 of the Nine Mile Point Unit 2 Final Safety Analysis Report will be revised to identify the following Type A variables.

   1. Containment hydrogen concentration (2) (same as variables C11a and b)

   2. Containment oxygen concentration (2) (same as variables C12a and b)

   3. Reactor vessel pressure (2) (same as variables B6a and b)

   4. Reactor vessel level (4) (same as variables B4a, b, c and d)

   5. Suppression pool water temperature (8) (same as variables D6a and b)

   6. Drywell atmosphere temperature (18) (same as variables D7a and b)

   7. Drywell atmosphere pressure (2) (same as variables D4a and b)

2. Neutron flux - The applicant should provide redundant Class 1E power sources for this instrumentation; the applicant should show that the source and intermediate ranges have sufficient overlap.

   Response

   The source of power (see Nine Mile Point Unit 2 Final Safety Analysis Report Figure 8.3-10) for Source Range Monitors (SRMs) and Intermediate Range Monitors (IRMs) originates from reliable normal dc sources. Normal supply originates from stub buses (2NJS-US6 and US5) to 24/48-V dc distribution panels (2BWS-PNL300A and 2BWS-PNL300B). A normal power source feeds two battery chargers per division that service the 24/48-V dc distribution panel(s) and maintain the charge on two 24-V dc batteries. The batteries are available to service the 24/48-V dc distribution panels on loss of normal power.

   The power source for Low Power Range Monitor (LPRM) groups and Average Power Range Monitor (APRM) channels is from the RPS/UPS channelized Divisions 1 through 4. This power is fed to RPS buses by means of a UPS which has normal, alternate and battery backup sources. Power sources are channelized RPS Divisions 1 through 4. The power distribution system for this instrumentation is described in detail in Section 8.3.1.1.3 of the Final Safety Analysis Report.

   It is Niagara Mohawk's determination that this design provides reliable power sources.

Although the power supplies are classified as nonsafety-related, this does not impair the ability of the neutron monitoring instrumentation to perform its required detection and trip functions. The trip function is configured to trip and initiate a scram on loss of power. The instrumentation is seismically and environmentally qualified, so the trip function is ensured on loss of power.

The operating ranges of the source range (SRM) and intermediate range (IRM) devices are as follows:

SRM = 1×10^3 to 1.5×10^9 nv
IRM = 1×10^8 to 1.5×10^{13} nv

The overlap of the ranges is 1×10^8 to 1.5×10^9 nv.

Additionally, there is a typographical error in Table 421.36-1. The lower end of the IRM range should be 4.0×10^{-5} percent power. This will be corrected in the next Final Safety Analysis Report amendment.

3. Reactor coolant system soluble boron concentration - The applicant should identify the range of the instrumentation being supplied for this variable.

Response

The range is 50 to 2,000 ppm boron in solution. This information will be incorporated in the next Final Safety Analysis Report update.

4. Coolant level in reactor - The applicant should identify the remainder of this instrumentation in accordance with Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations identified.

Response

Conformance to Regulatory Guide 1.97, Revision 3, is accomplished by the use of two transmitters per division, one fuel zone and one wide range. As noted in Section 3.3.3 of your November 15, 1985 letter, the wide range transmitters were omitted from the table and will be added in the next Final Safety Analysis Report amendment.

The wide range transmitters (2ISC*LT9C and D) are calibrated to monitor the 375.70 in. level, which is inside the fuel zone transmitter range, to the 585.70 in. level which is 62.3 in. below the centerline of the main steam lines at 648 in.

This range is considered to meet the intent of the regulatory guide which is to restore and maintain reactor pressure vessel water level to ensure adequate core cooling.

5. Drywell pressure - The applicant should provide independent Class 1E power sources for these instrument channels.

Response

The two instrument channels are powered from separate Class 1E sources. Table 421.36-1 is incorrect and will be corrected in a future Final Safety Analysis Report amendment.

6. Drywell sump level - The applicant should provide instrumentation for this variable.

Response

See Item 7.

7. Drywell drain sumps level - The applicant should provide instrumentation for this variable.

Response (Items 6 and 7)

The drywell sump level instruments provide indication of identified and unidentified leakage during normal operating conditions. The instrumentation, which is nonsafety grade (Category 3), is located on drain tanks in the secondary containment, outside of the drywell. Under accident conditions, these drain tanks are automatically isolated from the primary containment to prevent the escape of any post-accident reactor fluid from the drywell. In this situation, the drywell sump level indication is no longer meaningful and thus serves no post-accident safety function.

Other instrumentation is available to identify leakage into the drywell. This includes drywell pressure, drywell temperature, and containment radiation. These instruments meet the Category I requirements of Regulatory Guide 1.97.

8. Radiation level in circulating primary coolant - The applicant should supply the recommended instrumentation and the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations from the regulatory guide, and justify those deviations.

Response

This instrumentation is not provided at Nine Mile Point Unit 2.

Justification

The usefulness of information obtained by monitoring the radiation level in the circulating primary coolant, in terms of helping the operator in his efforts to prevent and mitigate accidents has not been substantiated. The particular planned operator action to be taken based on monitoring this variable is not specified in the current draft of the Emergency Procedures. The critical actions taken to prevent and mitigate a gross breach of fuel cladding are to shutdown the reactor and maintain water level. Monitoring primary coolant radioactivity has no influence on either of these actions. The purpose of this monitor falls in the cateogry of "information that the barriers to release of radioactive material are being challenged" and

"identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." Additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables, have not been identified.

Regulatory Guide 1.97 specifies measurement of the radioactivity of the circulating primary coolant as the key variable in monitoring fuel cladding status during isolation of the nuclear steam supply system (NSSS). The words "circulating primary coolant" are interpreted to mean coolant, or a representative sample of such coolant, that flows past the core. A basic criterion for a valid measurement of the specified variable is that the coolant being monitored is coolant that is in active contact with the fuel, i.e., flowing past the failed fuel. Monitoring the active coolant (or a sample thereof) is the dominant consideration. The post-accident sampling system (PASS) provides a representative sample which can be monitored.

The concern of Regulatory Guide 1.97 assumes a situation in which the NSSS is isolated and the reactor is shutdown. This assumption is justified because the monitors in the off-gas system and main steam tunnel provide reliable and accurate information on the status of fuel cladding when the plant is not isolated. Further, the PASS, once activated, provides an accurate status of coolant radioactivity and hence, cladding status. In the interim between NSSS isolation and operation of the PASS, monitoring of the primary containment radiation and hydrogen levels provides information on the status of the fuel cladding.

Present emergency procedures provide that once initial core damage is estimated using information obtained from the analysis of PASS samples, the estimate is confirmed using containment hydrogen analysis, containment high-range radiation monitoring, water level indications, and Sr, Ba, La, and Ru analyses. Therefore, no Type C Category I instrumentation is provided to measure the subject variable.

The Niagara Mohawk position agrees with the BWR Owners Group position on this variable.

9. Analysis of primary coolant - The applicant should identify the range of the instrumentation being supplied for this variable.

Response

The Instrument range is 10^{-6} to 10^{1} Ci/gm and will be incorporated in the next Final Safety Analysis Report update.

10. Radiation exposure rate - The applicant should show that the ranges encompass the expected radiation levels in their locations.

Response

Access is not required in any area of secondary containment to service safety-related equipment in a post-accident situation. When accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere.

Area monitors provided in areas outside secondary containment where access may be required post-accident have ranges that envelop the dose rates expected in these areas at the time access is required.

11. Residual heat removal heat exchanger outlet temperature - Environmental qualification should be addressed in accordance with 10CFR50.49.

Response

The instrumentation for this variable is Category I and environmentally qualified to 10CFR50.49. It is in compliance with the requirement for Regulatory Guide 1.97, Category 2 instrumentation. Table 421.36-1 of the Final Safety Analysis Report is incorrect and will be corrected in a future amendment.

12. Cooling water temperature to engineered safety feature system components - The applicant should justify the deviation in range.

Response

The temperature range of 2SWP*TE31A and B complies with the intent of Nuclear Regulatory Commission Regulatory Guide 1.97, Revision 3. The intent of the regulatory guide is to ensure that instrument ranges are selected so that the instrument will always be on scale. 2SWP*TE31A and B are located on the service water supply header and are used to monitor service water supply temperature. The temperature range of service water instrument is based upon the range for Lake Ontario, which normally varies from 38°F to 77°F and which is well within the instrument range of 32°F to 130°F.

13. Secondary containment area radiation - The applicant should supply the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations; environmental qualification should be addressed in accordance with 10CFR50.49.

Response

Environmentally qualified area monitors with ranges of 10^{-1} to 10^{4} R/hr are not provided in secondary containment at Nine Mile Point Unit 2.

Justification

The use of local area radiation monitors to detect breach or leakage through primary containment penetrations is inappropriate. In general, radiation levels in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in emergency core cooling system (ECCS) piping. Localized hot spots due to piping sources and primary containment penetrations and hatches will provide ambiguous indications. Breach of primary containment will be detected by the reactor building exhaust gaseous effluent monitor prior to the reactor building isolation and the noble gas channel of the main stack gaseous effluent monitor following the isolation of the reactor building. Therefore, these area monitors are not necessary and are not implemented at Nine Mile Point Unit 2.

In the long term, accessibility to the secondary containment will be reestablished using a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere.

14. Noble gas, radwaste vent - The applicant should either provide the recommended range or justify the use of the lesser range.

Response

See Item 15.

15. Noble gas, common plant vent - The applicant should either provide the recommended range or justify the use of the lesser range.

Response (Items 14 and 15)

The noble gas channels of the gaseous effluent monitors for the radwaste/reactor building vent and the main stack release points have design ranges of 10^{-6} to 10^{4} uCi/cc, which meet Regulatory Guide 1.97 requirements.

16. Plant and environs radiation - The applicant should identify the ranges of this instrumentation and show that the ranges are adequate.

Response

Two types of portable radiation detection and instrumentation are provided to monitor the plant and environs. An ion chamber detector with a range of 10^{-3} to 50 R/hr is used for low-level gamma and beta radiation monitoring. A Geiger-Muller Teletector type detector with a range of 10^{-2} to 10^{3} R/hr is used for high level gamma radiation monitoring. With a combined range of 10^{-3} to 10^{3} R/hr, these instruments have adequate range to envelop the dose rates expected outside the plant buildings after an accident.

17. Accident sampling (primary coolant, containment air and sump) - The applicant should provide the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations from the regulatory guide, and justify those deviations.

Response

1. Instrument range - Analysis range is given in Table II.B.3-2 (See Section 1.10, Table II.B.3-2 of Nine Mile Point Unit 2 Final Safety Analysis Report). The ranges meet or exceed requirements of Regulatory Guide 1.97, within instrument limitations, with the exception of the dissolved gas sample analysis. The ranges given for dissolved gas analysis were approved by the Nuclear Regulatory Commission in a letter to General Electric (letter from W. Johnston [Nuclear Regulatory Commission] to G. Sherwood [General Electric] dated July 17, 1984).

2, 3, 4, 6. Environmental qualification, seismic qualification, quality assurance, and power supply - These have been addressed in Table 421.36-1 and meet the requirements of Regulatory Guide 1.97.

5. Redundancy and sensor locations - This is not applicable, since analysis is done in a chemistry laboratory on grab samples. Regulatory Guide 1.97 has no specific provision.

7. Display location - This is not applicable, since analysis is done in a chemistry laboratory, the display is on each individual instrument. This meets the requirements of Regulatory Guide 1.97.

18. Primary containment isolation valve position - The applicant should provide justification for the exemption of instrumentation for the traversing incore probe system isolation valves.

Response

The traversing incore probe (TIP) system isolation valves consist of ball valves, operated when the probe is out of the guide tube, and shear valves manually operated if the probe is in the guide tube.

The TIPs are normally withdrawn and the ball valves are closed. If an event occurs while the TIP is inserted into the core and the TIP should fail to retract, the shear valve can be operated manually to provide the necessary containment isolation.

These valves are classified nonessential and are provided with non-Class 1E automatic isolation signals and power and as such, cannot meet Regulatory Guide 1.97, Category 1 or 2 requirements. (For further explanation, refer to the Nine Mile Point Unit 2 Final Safety Analysis Report Section 1.10, TMI action item II.E.4.2 concerning the Containment Isolation Dependability.)

Additionally, any leakage through this line has been incorporated in the radiological LOCA analysis of Chapter 15.6.5 of the Nine Mile Point Unit 2 Final Safety Analysis Report.

19. Please verify that Category I instrumentation is or will be provided for neutron flux instrumentation (from W. R. Butler letter of October 15, 1985). (Table 421.36-1 of Final Safety Analysis Report commits to environmentally and seismically qualified equipment.)

Response

Neutron flux measurement devices located in harsh environment areas are environmentally and seismically qualified for the anticipated environments. The environmental qualification under harsh environment conditions is for a limited time, but the time is sufficient to perform the detection, mitigation, and monitoring functions required of the instrumentation. Instrumentation located in mild environment areas is seismically qualified.

~~Harold Denton~~ V. Hauck LP-25

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EISENHUT,D.G. Division of Licensing

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	NRR/DE/EQB	13	2	2	NRR/DE/GB	28	2	2
	NRR/DE/MEB	18	1	1	NRR/DE/MTEB	17	1	1
	NRR/DE/SAB	24	1	1	NRR/DE/SGEB	25	1	1
	NRR/DHFS/HFEB	40	1	1	NRR/DHFS/LQB	32	1	1
	NRR/DHFS/PSRB		1	1	NRR/DL/SSPB		1	0
	NRR/DSI/AEB	26	1	1	NRR/DSI/ASB		1	1
	NRR/DSI/CPB	10	1	1	NRR/DSI/CSB	09	1	1
	NRR/DSI/ICSB	16	1	1	NRR/DSI/METB	12	1	1
	NRR/DSI/PSB	19	1	1	NRR/DSI/RAB	22	1	1
	NRR/DSI/RSB	23	1	1	REG FILE	04	1	1
	RGN1		3	3	RM/DDAMI/MIB		1	0
EXTERNAL:	ACRS	41	6	6	BNL(AMDTS ONLY)		1	1
	DMB/DSS (AMDTS)		1	1	FEMA-REP DIV	39	1	1
	LPDR	03	1	1	NRC PDR	02	1	1
	NSIC	05	1	1	NTIS		1	1
NOTES:			1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR 56 ENCL 48

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

OCTOBER 5, 1984
(NMP2L 0169)

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

Re: Amendment 14 to Application to Operating License
Nine Mile Point Unit 2
Docket No. 50-410

In accordance with 10CFR50.30 (c) (1), 10CFR51.24 and your March 29, 1983 letter, enclosed are three originals and 60 copies of Amendment 14 to the Final Safety Analysis Report. These changes incorporate certain responses into the text of the Final Safety Analysis Report as appropriate. Also included are changes which have resulted from our continuing review of these documents.

Very truly yours,

C. V. Mangan
Vice President
Nuclear Engineering & Licensing

JM:ja
Enclosure
xc: Project File (2)

B001

3/60

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

C. V. Mangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 24th day of September, 1984.

Christine Austin
Notary Public in and for
Onondaga County, New York

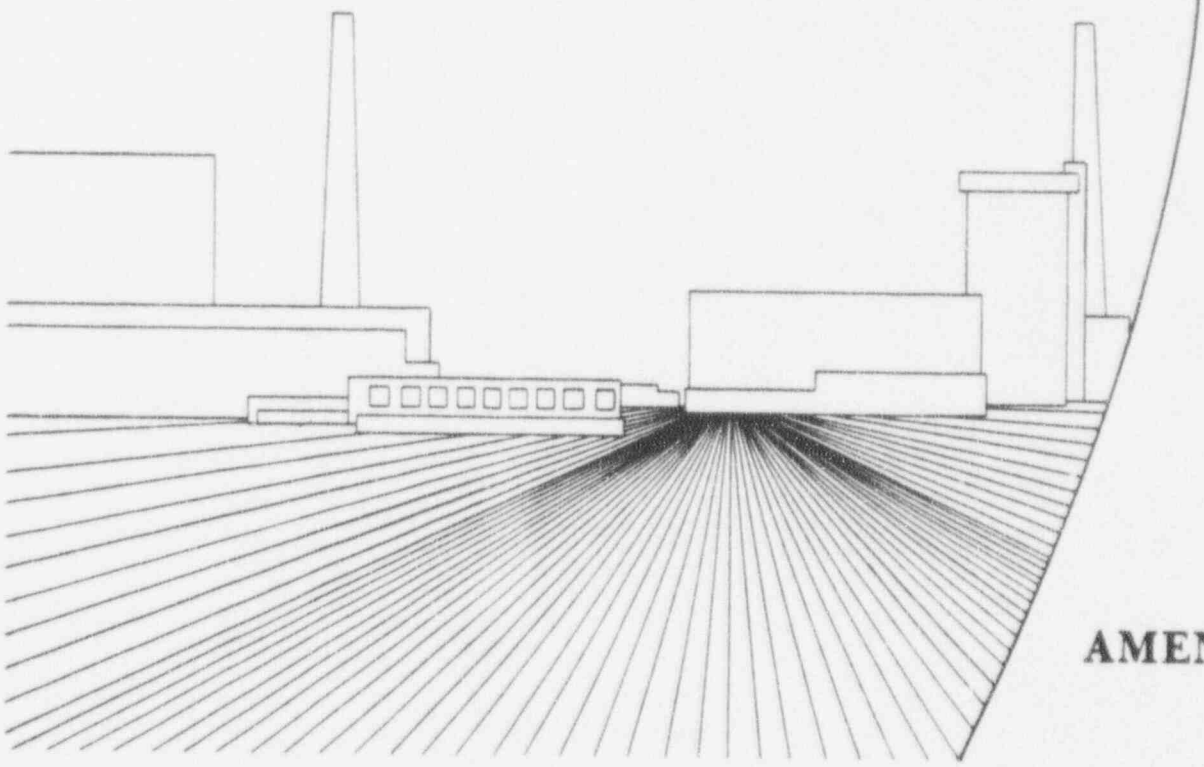
My Commission expires:

CHRISTINE AUSTIN
Notary Public in the State of New York
Qualified in Onondaga Co. No. 4787687
My Commission Expires March 30, 19[illegible]

FINAL SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION – UNIT 2

NIAGARA MOHAWK



AMENDMENT 14

20PP

QUESTION F421.36 (7.5)

The NRC staff has recently issued Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" via Supplement 1 to NUREG-0737. This Reg. Guide revision reflects a number of major changes in post-accident instrumentation. Supplement 1 to NUREG-0737 includes specific Reg. Guide 1.97 implementation requirements for plants in the operating license review stage.

Provide a description of how the Nine Mile Point Unit 2 design conforms to the provisions of Reg. Guide 1.97, Revision 2. This description should be in the form of a table that includes the following information for each Type A, B, C, D, E variable shown in Regulatory Guide 1.97:

(1) instrument range

(2) environmental qualification (as stipulated in guide or state criteria)

(3) seismic qualification (as stipulated in guide or state criteria)

(4) quality assurance (as stipulated in guide or state criteria)

(5) redundancy and sensor(s) location(s)

(6) power supply (e.g., Class 1E, non-Class 1E, battery backed)

(7) location of display (e.g., control room board, SPDS, chemical laboratory

Deviations from the guidance in Reg. Guide 1.97 should be explicitly shown, and supporting justification or alternatives should be presented.

RESPONSE

See Table 421.36-1. 14

TABLE 421.36-1

CONFORMANCE TO REGULATORY GUIDE 1.97

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
B13-D193	Power Rng Flux Level	B1a.	1	Core	0.5-125% pwr	Yes	Yes	II	Non-1E	P603	1
	Average Pwr Rng Flux Lvl	B1b.	1	N/A	0-125% pwr	No	No	II	Non-1E	P603	2
C51-N002A-H	Inter-mediate Rng Flux Level	B1c.	1	Core	40x10^{-9} - 12.6% Pwr	Yes	Yes	II	Non-1E	P603	-
C51-N001A-D	Source Rng Flux Level	B1d.	1	Core cps	0.1-1x10^{6}	Yes	Yes	II	Non-1E	P603	-
	Control Rod Position	B2	3	Core	Withdrawn or Scram	Yes	Yes	II	Non-1E	P603	-
	Rx Coolant Boron Conc	B3	3	Unit 1 H.P. Lab.	(ltr)	(ltr)	(ltr)	II	Non-1E	-	4
2ISC*LT13A/ B22-N044A	Reactor Vsl Level - A (Fuel Zone)	B4a.	1	Rx Bldg (Sec Containt)	230.64-430.69"	Yes	Yes	I	Div. 1	P601	5
2ISC*LT13B/ B22-N044B	Reactor Vsl Level - B (Fuel Zone)	B4b.	1	Rx Bldg (Sec Containt)	230.69-430.69"	Yes	Yes	I	Div. 2	P601	5
N/A N/A	Core Temperature	B5	1	-	-	-	-	-	-	-	6
2ISC*PT6A/ B22-N062A	Reactor Vsl Pressure-A	B6a.	1	Rx Bldg (Sec Containt)	0-1500 psig	Yes	Yes	I	Div. 1	P601	-
2ISC*PT6B/ B22-N062B	Reactor Vsl Pressure-B	B6b.	1	Rx Bldg (Sec Containt)	0-1500 psig	Yes	Yes	I	Div. 2	P601	-

TABLE 4.21.36-1 (Cont)

SNEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CMS*PT2A	Drywell Pressure-A	B7a.	1	Rx Bldg (Sec Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P601	-
2CMS*PT2B	Drywell Pressure-B	B7b.	1	Rx Bldg (Sec Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P898	-
2CMS*PT7A	Suppression Chamber Pressure-A	B7c.	1	Rx Bldg (Sec (Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P601	-
2CMS*PT7B	Suppression Chamber Pressure-B	B7d.	1	Rx Bldg (Sec Contmt)	0-150 psig	Yes	Yes	I	Div. 1	P898	-
See Note 7	Drywell Sump Level	B8	1	-	-	-	-	-	-	-	7
2CMS*PT1A	Primary Containment Pressure-A	B9a.	1	Rx Bldg (Sec Contmt)	-5 to +5 psig	Yes	Yes	I	Div. 1	P601	8
2CMS*PT1B	Primary Containment Pressure-B	B9b.	1	Rx Bldg (Sec Contmt)	-5 to +5 psig	Yes	Yes	I	Div. 2	P601/ P898	8
2AAS*MOV134, 135	Primary Containment Vlv Isolation - AAS	B10a1	1	N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2AAS*MOV136, 137	Primary Containment Vlv Isolation - ASS	B10a1	1	N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2CCP*MOV17A, B;124, 265	Primary Containment Vlv Isolation - CCP	B10b1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602/ P873	37

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CCP*MOV16A, B; 94A, B; 122, 273	Primary Containment Vlv Isolation - CCP	B10b2	1	N/A	N/A	Yes	Yes	I	Div. 2	P602/ P873	37
2CMS*SOV24A, C;26A,C;32A; 33A;34A;35A; 60A,B;63A,B	Primary Containment Isolation - CMS	B10c1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2CMS*SOV14B, D;26B,D;32B; 33B;34B;35B; 61A,B;63A,B	Primary Containment Isolation - CMS	B10c2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-
2CPS*AOV104, 105,110,111 *SOV119,120	Primary Containment Isolation - CPS	B10d1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2CPS*MOV106, 107,108,109 *SOV121,122	Primary Containment Isolation - CPS	B10d2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-
2CSH*AOV108 *MOV105,107 111,118 E33-F005 -F012,F004, F023,F015	Primary Containment Isolation - CSH	B10e	1	N/A	N/A	Yes	Yes	I	Div. 3	P601	-
2CSL*AOV101 *MOV104,112 E21-F006 -F005,F001	Primary Containment Isolation - CSL	B10f	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-
2DER*MOV120, 131	Primary Containment Isolation - DER	B10q1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2DEP*MOV119, 130	Primary Containment Isolation - DEP	B10q2.	1	N/A	N/A	Yes	Yes	I	Div. 2	P873	-
2DFR*MOV120, 139	Primary Containment Isolation - DFR	B10h1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2DFR*MOV121, 140	Primary Containment Isolation - DFR	B10h1	1	N/A	N/A	Yes	Yes	I	Div. 2	P873	-
2FPW*SOV218, 220	Primary Containment Isolation - FPW	B10j1	1	N/A	N/A	Yes	Yes	I	Div. 1	P849	36
2FPW*SOV219, 221	Primary Containment Isolation - FPW	B10j2	1	N/A	N/A	Yes	Yes	1	Div. 2	P849	36
2FWS*MOV21A, B 822-F065A,B	Primary Containment Isolation - FWS	B10k1	1	N/A	N/A	Yes	Yes	I	Div. 1	P603	-
2FWS*AOV23A, B 822-F032A,B	Primary Containment Isolation - FWS	B10k2.	1	N/A	N/A	Yes	No	II	Non-1E	P603	9
2HCS*MOV1A, 2A,3A,4A, 5A,6A	Primary Containment Isolation - HCS	B10L1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2HCS*MOV1B, 2B,3B,4B, 5B,6B	Primary Containment Isolation - HCS	B10L2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2IAS*SOV164, 166,167,168	Primary Containment Isolation - IAS	B10m1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601/ P851	38
2IAS*SOV165, 180,184,185	Primary Containment Isolation - IAS	B10m2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601/ P851	38
2ISC*MOV121, 122,126,136 143,164 *AOV156,157 E51-F064,F068, F013, F031, F019, F080, E51-F065,F066	Primary Containment Isolation - ICS	B10n1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-
2ICS*MOV128, 148,170 E51-F063, F086,F076	Primary Containment Isolation - ICS	B10n2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601	-
2LMS*SOV153, 157	Primary Containment Isolation - LMS	B10p1	1	N/A	N/A	Yes	Yes	I	Div. 1	P873	-
2LMS*SOV152, 156	Primary Containment Isolation - LMS	B10p2	1	N/A	N/A	Yes	Yes	I	Div. 2	P875	-
2MSS*MOV112, 20B B22-P019,N/A	Primary Containment Isolation - MSS	B10q1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602	-
2MSS*HIV7A, B,C,D B22-F028A,B, C,D	Primary Containment Isolation - MSS	B10q2	1	N/A	N/A	Yes	Yes	I	RPS Div. 1	P602	-

TABLE 821.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2MSS*MOV111 B22-F016	Primary Containment Isolation - MSS	B10q3	1	N/A	N/A	Yes	Yes	I	Div. 2	P602	-
2MSS*HYV6A, B,C,D B22-F022A,B, C,D	Primary Containment Isolation - MSS	B10q4	1	N/A	N/A	Yes	Yes	I	RPS Div 2	P602	-
C51-J004A,B, C,D,E	Primary Containment Isolation - NMS	B10r	1	N/A	N/A	Yes	No	II	Non-1E	P607	10
2RCS*SOV65A, B;66A,B;67A, B 2RCS*SOV104 B35-F020	Primary Containment Isolation - RCS	B10s1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602	-
2RCS*SOV79A, B;80A,B;81A, B;82A,B 2RCS*SOV105 B35-F019	Primary Containment Isolation - RCS	B10s2	1	N/A	N/A	Yes	Yes	I	Div. 2	P602	-
2RHS*MOV1A, 15A,16A,24A, 25A,26A,27A, 30A,33A,39A, 40A,67A,104, 113 B12-F004a,F016A,F041A, F042A,F017A,F074A,F079A F105A,F027A,F050A,F053A F099A,F023,F008	Primary Containment Isolation - RHS	B10t1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-

TABLE 7A21.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2RHS*MOV18, 1C;15B;16BC; 24A,B;25B; 26B;27B;30B; 33B;34B;40B;61B;112 E12-F004B,C;F016B; F041BC;F042B,C;F017B;F074B; F073B;F105B;F037B;F050B; F059B;F099B;F009	Primary Containment Isolation - RHS	B10t2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601	-
2SAS*MCV160, 161	Primary Containment Isolation - SAS	B10u1		N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2SAS*MCV162, 163	Primary Containment Isolation - SAS	B10u2	1	N/A	N/A	Yes	Yes	I	Div. 1	P851	-
2SLS*MOV5A C41-F006A	Primary Containment Isolation - SLS	B10v1	1	N/A	N/A	Yes	Yes	I	Div. 1	P601	-
2SLS*MOV5B C41-F006B	Primary Containment Isolation - SLS	B10v2	1	N/A	N/A	Yes	Yes	I	Div. 2	P601	-
2WCS*MOV112, 200A G33-F004, F040	Primary Containment Isolation - WCS	B10v1	1	N/A	N/A	Yes	Yes	I	Div. 1	P602	-
2WCS*MOV102 G33-F001	Primary Containment Isolation - WCS	B10v2	1	N/A	N/A	Yes	Yes	I	Div. 2	P602	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
	Radioactive Concentration in Primary Coolant	C1	1	Unit 1 H.P. Lab	(ltr)	(ltr)	(ltr)	II	Non-1E	-	4
	Analysis of Primary Coolant Gamma Spectrum	C2	3	Unit 1 H.P. Lab	(ltr)	(ltr)	(ltr)	II	Non-1E	-	4
N/A N/A	Core Temperature	C3	1	-	-	-	-	-	-	-	6
See Note 11 See Note 11	Reactor Coolant System Pressure	C4	1	-	-	-	-	-	-	-	11
See Note 12	Primary Containment Area Radiation	C5	3	-	-	-	-	-	-	-	12
N/A N/A	Drywell Drain Sumps Level	C6	1	-	-	-	-	-	-	-	13
2CMS*LT11A	Suppression Pool Water Level (Narrow Rng)	C7a	1	Rx Bldg (Sec Contmt)	197-202 ft	Yes	Yes	I	Div. 1 P601		14
2CMS*LT11B	Suppression Pool Water Level (Narrow Rng)	C7b	1	Rx Bldg (Sec Contmt)	197-202 ft	Yes	Yes	I	Div. 2 P898		14

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CMS*LT9A	Suppression Pool Water Level (Wide Rng)	C7c	1	Rx Bldg (Sec Contmt)	192-217 ft	Yes	Yes	I	Div. 1	P601	14
2CMS*LT9B	Suppression Pool Water Level (Wide Rng)	C7d	1	Rx Bldg (Sec Contmt)	192-217 ft	Yes	Yes	I	Div. 2	P898	14
See Note 15	Drywell Pressure	C8	1	-	-	-	-	-	-	-	15
See Note 16	Reactor Coolant System Pressure	C9	1	-	-	-	-	-	-	-	16
See Note 17	Primary Containment Pressure	C10	1	-	-	-	-	-	-	-	17
2CMS*AIT6A	Containment Hydrogen Concentration - A	C11a	1	Rx Bldg North Aux Bay	0-30%	Yes	Yes	I	Div. 1	P601	-
2CMS*AIT6B	Containment Hydrogen Concentration - B	C11b	1	Rx Bldg South Aux Bay	0-30%	Yes	Yes	I	Div. 2	P898	-
2CMS*AIT71A	Containment Oxygen Concentration - A	C12a	1	Rx Bldg North Aux Bay	0-10%	Yes	Yes	1	Div. 1	P601	-
2CMS*AIT71B	Containment Oxygen Concentration - B	C12b	1	Rx Bldg North Aux Bay	0-10%	Yes	Yes	I	Div. 2	P898	-

TABLE 421.36-1 (Cont)

SNEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2EMS-CAB170	Containment Effluent Radioactivity	C13	3	Main Stack Enclosure	Isotopic 10^{-7}-10^{5} uci/cc	No	Yes	II	Non-1E	P882	39
2RMS-CAB180 -	Effluent Radioactivity	C14	2	Turb Bldg Turb Oper Floor	Isotopic 10^{-7}-10^{5} uci/cc	No	Yes	II	Non-1E (UPS)	P882	39
2FWS-FT1A,B C33-N001A,B	Main Feedwater Flow -A,B		3	Turb Bldg	Turb Bldg 0-8.5 lbs/hr (each)	No	No	II	Non-1E	P603	
2CWS-LT8A,B	Condensate Storage Tk Level - A, B	D2	3	Cond Stor TK1A,TK1B	0-500 K gal (each)	No	No	II	Non-1E	P851	-
2RHS*FT64A	Suppression Chamber Spray Header Flow - A	D3a	2	Rx Bldg (Sec Contmt)	0-4500 gpm	Yes	Yes	I	Div. 1	P601	-
2RHS*FT64B	Suppression Chamber Spray Header Flow - B	D3b	2	Rx Bldg (Sec Contmt)	0-4500 gpm	Yes	Yes	I	Div. 2	P601	-
See Note 18	Drywell Pressure	D4	2	-	-	-	-	-	-	-	18
See Note 19	Suppression Water Level (Weir Well)	D5	2	-	-	-	-	-	-	-	19
2CMS*TE67A, 68A,69A,70A	Suppression Pool Water Temp-A	D6a	2	Suppression Pool	50-250°F	Yes	Yes	I	Div. 1	P601	20
2CMS*TE67B, 68B,69B,70B	Suppression Pool Water Temp-B	D6b	2	Suppression Pool	50-250°F	Yes	Yes	I	Div. 2	P601/ P598	20

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2CMS*TE101 thru & inc 109	Drywell Atmos Temp - A	D7a	2	Drywell	0-400°F	Yes	Yes	I	Div. 1	P873	20
2CMS*TE116 thru & inc 124	Drywell Atmos Temp - B	D7b	2	Drywell	0-400°F	Yes	Yes	I	Div. 2	P875	20
2RHS*FT63A	Drywell Spray Header Flow - A	D8a	2	Rx Bldg (Sec Contmt)	0-7,950 gpm	Yes	Yes	I	Div. 1	P601	-
2RHS*FT63B	Drywell Spray Header Flow - B	D8b	2	Rx Bldg (Sec Contmt)	0-7,950 gpm	Yes	Yes	I	Div. 2	P601	-
See Note 19	Main Steam Line Isolation Valve Leakage Control System Press	D9	2	-	-	-	-	-	-	-	19
2SVV*ZT220 237	Primary Safety Relief Valve Position	D10a	2	Acoustic Sensor on Tail Pipe (18 total)	-	Yes	Yes	I	Div. 1	*SVV-PNL 140	-
2IAS*PT181	Primary Safety Relief Valve ADS Header Pressure-A	D10b	2	Rx Bldg (Sec Contmt)	0-250 psig	Yes	Yes	I	Div. 1	P601	-
2IAS*PT230, 231,232	Primary Safety Relief Valve - ADS Tank Pressure	D10c	2	Rx Bldg (Sec Contmt)	0-200 psig	Yes	Yes	I	Div. 1	P601	-

TABLE 7A1.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2IAS*PT186	Primary Safety Relief Valve - ADS Header Pressure-B	D10d	2	Rx Bldg (Sec Contmt)	0-250 psig	Yes	Yes	I	Div. 2	P601	-
2IAS*PT233, 234, 235, 236	Primary Safety Relief Valve - ADS Tank Pressure	D10e	2	Rx Bldg (Sec Contmt)	0-200 psig	Yes	Yes	I	Div. 2	P601	-
See Note 19	Isolation Condenser - Shell Side Water Level	D11	2	-	-	-	-	-	-	-	19
See Note 19	Isolation Condenser Valve Position	D12	2	-	-	-	-	-	-	-	19
2ICS*FT102 E51-N051	RCIC Flow	D13	2	Rx Bldg (Sec Contmt)	0-800 gpm	Yes	Yes	I	Div. 1	N/A	-
2CSH*FT105 E22-N056	HPCI/S Flow	D14	2	Rx Bldg (Sec Contmt)	0-1000 gpm	Yes	Yes	I	Div. 3	N/A	21
2CSL*FT107 E21-N051	LPCS Flow	D15	2	Rx Bldg (Sec Contmt)	0-10,000 gpm	Yes	Yes	I	Div. 1	N/A	-
2RHS*FT14A E12-N015A	LPCI Flow	D16a	2	Rx Bldg (Sec Contmt)	0-8,400 gpm	Yes	Yes	I	Div. 1	P601	-
2RHS*FT14B,C E12-N015B,C	LPCI Flow	D16b	2	Rx Bldg (Sec Contmt)	0-8,400 gpm	Yes	Yes	I	Div. 2	P601	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
2SLS*FT113 E41-N007	SLCS Flow	D17	2	Rx Bldg (Sec (Contmt)	0-86 gpm	Yes	Yes	I	Div. 1	P601	-
2SLS*LT103 C41-N001	SLCS Storage Tank Level	D18	2	Rx Bldg (Sec Contmt)	0-10,000 Gal	Yes	Yes	I	Div. 1	P601	-
See Note 22 See Note 22	RHR System Flow	D19	2	-	-	-	-	-	-	-	22
2PHS*TE13A E12-N027A	RHR Heat Exchanger Outlet Temp - A	D20a	2	Rx Bldg (Sec Contmt)	0-600°F	No	No	II	Non-1E	P601	-
2PHS*TE139 E12-N027B	RHR Heat Exchanger Outlet Temp - B	D20b	2	Rx Bldg (Sec Contmt)	0-600°F	No	No	II	Non-1E	P601	-
2SWP*TE31A	Cooling Water Temp to ESF System Components - A	D21a	2	Screen-well Bldg	35-130°F	Yes	Yes	I	Div 1	P601	20
2SWP*TE31B	Cooling Water Temp to ESF System Components - B	D21b	2	Screen-well Bldg	35-130°F	Yes	Yes	I	Div. 2	P601	20
2SWP*FT13A E12-N007A	Cooling Water Flow to ESF System Components - A	D22a	2	Rx Bldg (Sec Contmt)	0-10,000 gpm	Yes	Yes	I	Div. 1	P601	23
2SWP*FT138 E12-N007B	Cooling Water Flow to ESF System Components - B	D22b	2	Rx Bldg (Sec Contmt)	0-10,000 gpm	Yes	Yes	I	Div. 2	P601	23

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes	
2SWP*PT76A	Cooling Water Flow to ESF System Components - Div. 1 psi	D22c	2	Diesel Gen. Bldg	0-860 gpm	Yes	Yes	I	Div. 1	P852	2a	
2SWP*PT76B	Cooling Water Flow to ESF System Components - Div. 2 DSL	D22d	2	Diesel Gen. Bldg	0-860 gpm	Yes	Yes	I	Div. 2	P852	2a	
2SWP*PT535	Cooling Water Flow to ESF System Components - Div. 3 DSI	D22e	2	Diesel Gen. Bldg	0-650 gpm	Yes	Yes	I	Div. 2	P852	2a	
2LWS-2A,B,C; 26A,B;276; 280	High Radio-activity Liquid Tank Level	D23	3	Radwaste Bldg	0-100%	No	No	II	Non-1E	LWCS Computer Graphics	-	14
2HVP*AOD1A, 6A,9A,10A	Emergency Ventilation Damper Position	D24a	2	Rx Bldg (Sec Contmt)	N/A	Yes	Yes	I	Div. 1	P870	-	
2HVR*AOD1B, 6B,9B,10B	Emergency Ventilation Damper Position	D24b	2	Rx Bldg (Sec Contmt)	N/A	Yes	Yes	I	Div. 2	P871	-	
N/R	Status of Stdby Pwr Sources - Battery Voltage - 1	D25a	2	2BYS* SWG002A	0-150 vdc	Yes	Yes	I	Div. 1	P852	-	
2BYS*E/E1A	Status of Stdy Pwr Sources - Battery Current - 1	D25b	2	2BYS* SWG003A	-2000 to 3000 amps	Yes	Yes	I	Div. 1	P852	-	

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
N/R	Status of Stdby Pwr Sources - Battery Voltage - 2	D25c	2	2BYS* SWG003B	0-150 vdc	Yes	Yes	I	Div. 2	P852	-
2BYS*E/E18	Status of Stdby Pwr Sources - Battery Current - 2	D25d	2	2BYS* SWG002B	-2000 to +2000 amps	Yes	Yes	I	Div. 2	P852	-
N/R	Status of Stdby Pwr Sources - Battery Voltage - 3	D25e	2	2EGS* PNL002	0-150 vdc	Yes	Yes	I	Div. 3	P852	-
2BYS*E/E101	Status of Stdby Pwr Sources - Battery Current - 3	D25f	2	2EGS* PNL002	-100 to +100 amps	Yes	Yes	I	Div. 3	P852	-
N/R	Status of Stdby Pwr Sources - UPS Voltage - A	D25g	2	2YBS* UPS2A	0 to 120 vac	Yes	Yes	I	Div. 1	PP52	-
N/R	Status of Stdby Pwr Sources - UPS Current - A	D25h	2	2YBS* UPS2A	0 to 250 amps	Yes	Yes	I	Div. 1	N/A	-
N/R	Status of Stdby Pwr Sources - UPS Voltage - B	D25j	2	2UBS* UPS2B	0 to 120 vac	Yes	Yes	I	Div. 2	P852	-
N/R	Status of Stdby Pwr	D25k	2	2UBS* UPS1B	0 to 250 amps	Yes	Yes	I	Div. 2	N/A	-

TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
	Sources - UPS Current - B										
N/R	Status of Stdby Pwr Sources - 600V Swgr Voltage	D25l	2	2EJS*AS1	0 to 750 vac	Yes	Yes	I	Div. 1	2EJS*US1	25
N/R	Status of Stdby Pwr Sources - 600V Swgr Current	D25m	2	2EJS* X1A,B	0 to 3,000 amps	Yes	Yes	I	Div. 1	N/A	26
N/R	Status of Stdby Pwr Sources - 600V Swgr Voltage	D25n	2	2EJS* US3	0 to 750 vac	Yes	Yes	I	Div. 2	2EJS*US3	25
N/R	Status of Stdby Pwr Sources - 600V Swgr Current	D25p	2	2EJS* X3A,B	0 to 3,000 amps	Yes	Yes	I	Div. 2	N/A	26
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Voltage	D25q	2	Swgr	0 to 4.16 kv	Yes	Yes	I	Div. 1	P852	25
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Current	D25r	2	Swgr	0-1000 amp Diesel feed or 1,500 amp normal and alt feeds	Yes	Yes	I	Div. 1	P852	27

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TABLE 421.36-1 (Cont)

SWEC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Voltage	D25s	2	Swgr	0 to 4.16 kv	Yes	Yes	I	Div. 2	P852	25
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Current	D25t	2	Swgr	0-1,000 amp Diesel Feed 0-1,500 amp normal & alt feeds	Yes	Yes	I	Div. 2	P852	27
N/R	Status of Stdby Pwr Sources - 4 kV Swgr Voltage	D25u	2	Swgr	0 to 4.16 kv	Yes	Yes	I	Div. 3	P852	25
N/R	Status of Stdby Pwr Sources - 4kV Swgr Current	D25v	2	Swgr	0-600 amp Diesel Feed 0-1,500 amp normal & alt feeds	Yes	Yes	I	Div. 3	P852	27
See Note 28	Status of Stdby Pwr Sources - Air for ADS	D25w	2	-	-	-	-	-	-	-	28
2RMS*RE1A,C	Primary Containment Area Radiation High Rng	E1a	1	Drywell	1-10^7 R/hr	Yes	Yes	I	Div. 1	P880	-
2RMS*RE1B,D	Primary Containment Area Radiation High Rng	E1b	1	Drywell	1-10^7 R/hr	Yes	Yes	I	Div. 2	P880	-
See Note 29	Secondary Containment Area Radiation	E2	2	-	-	-	-	-	-	-	29
See Note 30	Vital Area Radiation	E3	3	See Note 30	See Note 30	NO	No	II	Non-1E	Radn Computer	30,31

TABLE 4Z1.36-1 (Cont)

SWFC/ GE-NED I.D.#	Parameter Description	Reg. Guide 1.97, Rev. 3 Parameter Variable	Parameter Classification	Sensor Location	Sensor Instr. Range	Qualification Seismic	Qualification Environmental	QA Class	Power Supply	Display Location	Notes
	Monitors									Graphics	
See Note 32	Effluent Radiation Released from Plant	E4	2	-	-	-	-	-	-	-	32
See Note 33	Environs Radiation and Radio-activity	E5	3	-	-	-	-	-	-	-	31,33
See Note 34	Meteorology	E6	3	-	-	-	-	-	Non-1E	P882	31,34
2SSP-IPNL101 D24-D007	Accident Sampling Primary Coolant and Containment Air	E7	3	Radwaste Sample Room	N/A	No	No	II	Non-1E	-	4,35

07-465 91A

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Docket No. 50-220

AUG 0 8 1990

Mr. Lawrence Burkhardt III
Executive Vice President,
Nuclear Operation
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Burkhardt:

SUBJECT: RESPONSE TO NRC GENERIC LETTER 89-06 ON THE SAFETY PARAMETER DISPLAY SYSTEM FOR NINE MILE POINT NUCLEAR STATION, UNIT 1 (MPA F-072, TAC NO. 73678)

NRC Generic Letter (GL) 89-06, dated April 12, 1989, requested you to provide certification regarding the implementation of a Safety Parameter Display System (SPDS) at your facility. The GL and its attachment, NUREG-1342, provided clarification of the requirements for an acceptable SPDS as originally defined in NUREG-0737, Supplement 1, issued January 1983. The GL further requested you to complete a checklist and take photographs of your SPDS and to retain these records for three years from the date of certification.

On July 11, 1989, Niagara Mohawk Power Corporation (the licensee) certified that the SPDS at Nine Mile Point Unit 1 met the requirements of NUREG-0737, Supplement 1, taking into account the information provided in NUREG-1342. This letter also identified that the EOPs are located away from the SPDS terminals. The staff concludes that this item is beyond the explicit scope of NUREG-0737, Supplement 1 for the SPDS and therefore does not require implementation to support our conclusion that the requirements of NUREG-0737, Supplement 1 have been met. Accordingly, based upon the certification provided in the NMPC letter of July 11, 1989, the NRC staff concludes that your facility has satisfactorily met all the requirements for an SPDS specified in NUREG-0737, Supplement 1. Therefore, staff review and licensee implementation of the SPDS are considered complete for your facility. Please contact the NRC project manager if you have any questions.

The reporting and/or recordkeeping requirements of this letter affect fewer than ten respondents; therefore OMB clearance is not required under PL 96-511.

Sincerely,

ORIGINAL SIGNED BY:

Robert E. Martin, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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cc: See next page

PDI-1*	PDI-1 *	BC:HFAB *	PDI-1 *	PDI-1
CVogan	DOudinot	JWermiel	REMartin	RACapra
/ /90	/ /90	/ /90	8/10/90	8/10/90

* See previous concurrence

Docket No. 50-220 AUG 0 8 1990

Mr. Lawrence Burkhardt III
Executive Vice President,
Nuclear Operation
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Burkhardt:

SUBJECT: RESPONSE TO NRC GENERIC LETTER 89-06 ON THE SAFETY PARAMETER DISPLAY SYSTEM FOR NINE MILE POINT NUCLEAR STATION, UNIT 1 (MPA F-072, TAC NO. 73678)

NRC Generic Letter (GL) 89-06, dated April 12, 1989, requested you to provide certification regarding the implementation of a Safety Parameter Display System (SPDS) at your facility. The GL and its attachment, NUREG-1342, provided clarification of the requirements for an acceptable SPDS as originally defined in NUREG-0737, Supplement 1, issued January 1983. The GL further requested you to complete a checklist and take photographs of your SPDS and to retain these records for three years from the date of certification.

On July 11, 1989, Niagara Mohawk Power Corporation (the licensee) certified that the SPDS at Nine Mile Point Unit 1 met the requirements of NUREG-0737, Supplement 1, taking into account the information provided in NUREG-1342. Based upon this certification, the NRC staff concludes that your facility has satisfactorily met all the requirements for an SPDS specified in NUREG-0737, Supplement 1. Therefore, staff review and licensee implementation of the SPDS are considered complete for your facility. Please contact the NRC project manager if you have any questions.

The reporting and/or recordkeeping requirements of this letter affect fewer than ten respondents; therefore OMB clearance is not required under PL 96-511.

Sincerely,

ORIGINAL SIGNED BY:

Robert E. Martin, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: See next page

PDI-1	PDI-1	BC/HFAB	PDI-1	PDI-1
CVogan	DOudinot	JWermiel	REMartin	RACapra
8/ /90	8/13/90	8/6/90	8/13/90	/ /90

Mr. L. Burkhardt III
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station,
Unit No. 1

cc:

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