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SUBJECT: Forwards addl info re 840402 submittal on Section 6 of Suppl 1 to NUREG-0737, ""Reg Guide 1,97-Application to Emergency Response Facilities requested in NRC 850613 ltr.Existing instrumentation meets needs for post-accident monitoring.

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NIAGARA MOHAWK

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October 18, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactor Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Subject: Request for Additional Information Concerning Niagara Mohawk Power Corporation's Submittal on Section 6 of Supplement 1 to NUREG-0737, "Regulatory Guide 1.97-Application to Emergency Response Facilities"

Dear Mr. Vassallo:

The attached document contains additional information comparing Nine Mile Point Unit 1 instrumentation to the requirements of Section 6.1 of Generic Letter 82-33 (Supplement 1 to NUREG-0737) regarding post accident monitoring for emergency response facilities, as requested in your June 13, 1985 letter. This comparison was initially made in our April 2, 1984 submittal and was based on the seven delineations contained in Section 6.2 (a) through (g) of this generic letter which, in turn, came from the guidance details contained in Regulatory Guide 1.97, Revision 2.

Relative to your request to identify any "incorrect assumptions or commitments beyond the intent of Niagara Mohawk's response," Section 6.1 allowed exceptions to its requirements, based on plant-specific design features, and these exceptions were the focus of our April 2, 1984 submittal. The purpose of that submittal was to show that the intent of the Section 6.2 requirements was met. However, except as detailed in that submittal, explicit compliance with all aspects of Regulatory Guide 1.97 should not be assumed.

Relative to the sub-tier supporting guidance of Regulatory Guide 1.97 (IEEE Standards, etc.), which are not specifically contained in Three Mile Island related documentation (i.e., Supplement 1 and the remainder of NUREG 0737), Nine Mile Point Unit 1 was reviewed as reported in the Technical Supplement to Petition for Conversion from Provisional Operating License to Full Term Operating License, July 1972. Our April 2, 1984 evaluations and comparisons were based on the status of the plant as established by the July 1972 submittal.

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The attachment contains supplemental information regarding the exceptions contained in our April 2, 1984 submittal. In addition, our other programs from Supplement 1 to NUREG 0737 covering the Emergency Operating Procedures, Detailed Control Room Design Review, Safety Parameter Display System, and Emergency Response Facilities, are now in the final phases. Evaluations aimed at determining the interactive effects (both analytical and physical) to program details and results are continuing to be carried out. The first two of the above programs have the strongest potential impact on instrumentation needs during post accident conditions and the existing instrumentation was found to be adequate for the scope of accident conditions covered, so far.

However, interactive evaluations will continue to be made until February, 1986, when the final analytical and formative steps in the above programs are carried out. The first of these steps is Emergency Operating Procedure changes to reflect Revision 4 of the generic Emergency Procedure Guidelines now being developed by the BWR Owners Group. Based on these final Emergency Operating Procedures, a re-review of the task analysis carried out in the Detailed Control Room Design Review will also be carried out; and the results of this will also be reviewed for impact on post accident instrumentation.

Based on current evaluations and developments, our conclusion continues to be that existing instrumentation adequately meets the needs for post accident monitoring and thus, the intent of Section 6 of Supplement 1 to NUREG-0737.

Sincerely,

NIAGARA MOHAWK POWER CORPORATION

CVMangan

C. V. Mangan
Senior Vice President

JLB:bd
Attachments

ATTACHMENT

Response to NRC Contractor's Request

for

Further Information Concerning

Niagara Mohawk's Submittal

on

Section 6 of Supplement 1 to NUREG-0737,

"Regulatory Guide 1.97 - Application to Emergency Response Facilities"

Nine Mile Point Unit #1

Niagara Mohawk Power Corporation

October 18, 1985

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[illegible]

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A. Introduction

The NRC contractor's report, transmitted by the NRC and received by Niagara Mohawk on June 20, 1985, contained 13 basic areas where exceptions are taken or questions asked about Niagara Mohawk's deviations in meeting the Section 6 requirements of Supplement 1 to NUREG-0737.

Section 6 pertains to the guidance of Regulatory Guide 1.97 concerning post accident instrumentation and has been coordinated by Niagara Mohawk with other sections of Supplement 1 to NUREG-0737. These other sections include the Emergency Operating Procedures (EOP's), Detailed Control Room Design Review (DCRDR), Safety Parameter Display System (SPDS) and Emergency Response Facilities. The EOP, DCRDR and SPDS programs involved substantial investigative/analytical steps, including:

- °The operator's recognition of response to a variety of postulated accident conditions,
- °A switch to fundamental symptomatic bases for action steps,
- °Human factor constraints, and
- °Other fundamental considerations that were substantially beyond the previously established Design Basis Accident concepts.

During these investigative/analytical steps, the ability of the operators to monitor particular parameters/boundary conditions was a key aspect - appropriate remedial responses depended upon this ability, both in sequence and timing.

Equipment changes and additions that resulted from these programs are in progress and will be substantially completed by the end of the spring 1986 refueling outage. Although final design detailing and analytical steps are not complete, in a few cases, no changes to instrumentation relative to the scope of coverage in Section 6 of Supplement 1 to NUREG-0737 have been identified, so far.

Some of the specific instrumentation listed in R.G. 1.97 and subject delineations of Section 6.2 are currently being covered by other regulatory programs. For example, 10 CFR 50.49 covers Environmental Qualification and other sections of NUREG-0737 cover other TMI requirements. These are noted in the NRC contractor's report and have been covered in this attachment by reference to the appropriate NMPC/NRC correspondence, including the current disposition of the subject, where needed to close out the item. Similarly, references to other regulatory actions which pertain to the issue involved have been included where they also add to appropriate resolution. An example is the seismic qualification program being proposed in the implementation plan for the NRC's USI A-46, described in section B.5. below.

Relative to the sub-tier supporting criteria of R.G. 1.97, concerning IEEE standards, QA program details, etc. NMPC's current programmatic status was identified in the April 2, 1984 submittal. Technical details are covered in the Technical Supplement to Petition for Conversion from Provisional Operating License to Full Term Operating License, July, 1972 and later documentation which developed these basic positions in more detail. (e.g., When the Q list was established.) These documents were assumed to remain the regulatory basis for these technical details and the current evaluations regarding this subtier R.G.1.97 guidance.

The 13 basic exception/question areas referred to at the beginning of this introduction will be discussed in Section B below in the sequence contained in the NRC contractor's report. The full text of the NRC contractor's statement from Section 3.3 will be repeated, and the specific summary conclusion statement from Section 4 will also be included. This will be followed by NMPC's response. This is a further elaboration of NMPC's original basis and justification for the deviation except for the identification of Type A variables (Item B.1 below). The latter was incomplete at the time of the April 2, 1984 submittal because the EOP's had not been completed.

There have also been some further developments or new information affecting the details contained in the April 2, 1984 submittal which will be pointed out in the section to which it pertains or noted at the end, if the subject was not covered in the NRC contractor's report. These resulted from NMPC's internal verification activities on the topic or are historical updatings.

B. Thirteen NRC Exception/Question Areas

1. Type A Variables

- NRC Contractor's Full Statement:

  "Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide information required to permit the control room operator to take specific manually controlled safety actions. As plant specific emergency operating procedures are not fully developed, the licensee has not defined the Type A variables. By the licensee's explicit commitment on conformance, we assume that all Type A variables will comply with Category 1 recommendations. However, the licensee should identify these Type A variables and verify that the instrumentation is Category 1."

- NRC Contractor's Summary Concern:

  "The licensee should identify Type A variables, and verify that they are Category 1."

- NMPC Elaboration:

  In the April 2, 1984 submittal, Niagara Mohawk evaluated 5 variables listed as "Proposed Type A Variables" from the BWR Owners Group work on R.G. 1.97, which is summarized in their report dated April 6, 1983. These were previously verified to meet the seven Category 1 criteria covered by Section 6.2 of Supplement 1 to NUREG-0737.

  Six other variables were listed as "Potential Type A Variables" in this BWROG report and these were recently evaluated by NMPC for applicability in the context of the recently prepared Emergency Operating Procedures (EOP's) for NMP-1. The results of this evaluation are summarized below.

  Two of these six variables (Condensate Storage Tank Level & Emergency Diesel Generator Load) do not apply because the consideration/action involved does not exist for NMP-1. (i.e. They are not a design feature of NMP-1.)

The third "Potential Type A Variable", O_2 or H_2 concentration, is incorporated for use in the EOP's and was previously listed and shown in the April 2, 1984 submittal to meet the Category 1 criteria as a Type C variable.

The fourth "Potential Type A Variable," Suppression Pool Pressure, is incorporated for use in the EOP's by using Primary Containment Pressure and a conservative relationship between the drywell and the wetwell during Design Basis Accidents. Primary Containment Pressure was also previously listed and shown to meet Category 1 criteria as a Type C variable, except for the sub-atmospheric portion of the required range. This exception will be eliminated as indicated in item B.4 below.

The fifth "Potential Type A Variable," Drywell Temperature, is incorporated for use in the EOP's and was previously listed and shown to meet Category 2 criteria for a Type D variable with an exception in the lower end of the range. This exception will be shown to be insignificant in item B.8 below. Category 1 criteria would also be substantially met, however, because additional instrumentation monitoring this parameter was installed as part of a recent plant modification to provide Remote Shutdown Panels (RSP's). This new Drywell Atmospheric Temperature monitoring instrumentation is redundant to the same monitors in the Control Room. The panel installation was classified as safety related and many of the components involved were purchased under this criteria even though they may not have been shown specifically to be handled this way.

The only other exception to Category 1 criteria for this parameter in the tables of the April 2, 1984 submittal related to the power supply. In this case, redundancy in power supplies and sensors is available if the temperature indicators on the Remote Shutdown Panels (RSP's) are considered. A dual element Thermocouple (T/C) was installed in one location of the drywell to supply the Control Room and one RSP, and the latter instrumentation is powered from a redundant class IE power supply (Reactor Protection System Bus 11). Another dual element T/C was installed in a different location of the drywell, also to supply the Control Room and the other RSP, but they are both powered from the same class IE power supply (Reactor Protection System Bus 12). The third T/C remained unchanged and is a single element supplying the Control Room, also powered from RPS 12. These RSP's are located in the Turbine Bldg. and are readily accessable from the Control Room, which is adjacent to the Turbine Bldg. Given Supplement 1 of NUREG-0737 allowances for design plant differences, this arrangement is considered adequate for meeting the intent of Category 1 criteria.

The sixth "Potential Type A Variable" involves the sump levels in the four reactor building corner rooms where the core and containment spray pumps are located and action may be needed to prevent flooding due to high energy line breaks outside the containment.

The Design Basis Accident involved is a break in one of three different sets of lines, the Main Steam Lines, Primary Coolant Cleanup System lines, and the Emergency Condenser Lines. All other lines penetrating the containment are either not connected to high energy sources or return to and are considered a part of the containment structural boundary. Since the Main Steam Lines do not penetrate the reactor building, they do not affect these sumps. Thus, the high energy line breaks that might affect the sumps would occur in the Primary Coolant Cleanup System and Emergency Condenser Lines. Leak/break detection instruments are provided for these systems which automatically generate signals to close isolation valves in these lines within 40 seconds. This instrumentation is also Reactor Protection System grade (i.e., Safety Related) and the automatic actuation of these isolation valves would indicate and annunciate in the Control Room.

These high energy lines are in locations different, vertically and laterally, from the corner rooms being considered. However, since these rooms are open to the Reactor Building, it is possible that water would run down and traverse through connecting floors and spaces and collect in the sumps. Per studies carried out as part of NMPC's Environmental Qualification (EQ) Program for NMP-1, the amount of water released within the maximum allowable closing time for the isolation valves would not rise to the level of the ECCS equipment in any of these corner rooms even if all the water accumulated in one room. (See item B.5. below for NMPC's EQ references.)

Besides the automatic actuation of the isolation valves, there are numerous other redundant indications to alert an operator to the accident condition. These include:

°High Level alarm from any of the four sumps which actuate an annunciator in Control Room,

°Local indicating lights for operation of each of the sump pumps,

°Area Radiation Monitors which actuate an annunciator in the Control Room,

°Continuous Air Monitors, which also alarm in the Control Room,

°Area Temperature Detectors, which also alarm in the Control Room, and,

°Indication in the Control Room of automatic initiation of the Emergency Ventilation System.

All of these indications would likely result from the high radiation in the primary cooling system water being released, or from the steam/water accumulating in the reactor building. These would be easily verified by operators in various parts of the Reactor Building since, for example, the corner rooms are

not isolated from the rest of the building. Although, these indicators would not meet Category 1 criteria in all cases, there is ample redundancy for various postulated situations.

Of course, further failures or degraded accident scenarios could be postulated which could cause additional water to accumulate in these rooms. Protective manual actions are called for in the new EOP's for such degraded possibilities, and these are appropriate for the conditions being considered.

The next question pursued in the evaluation of "Proposed" and "Potential" Type A variables was whether it was appropriate to classify any of these variables as Type A. The key distinction in the delineation of Type A variables is whether manual action is required to accomplish a safety related function during a Design Basis Accident. These variables (both "Proposed" and "Potential") were viewed as supplemental indicators, requiring close monitoring to be sure safety related actions are occurring properly, or to alert the operator of secondary manual actions needed to alleviate follow-on developing complications. However, none of them are required to accomplish safety-related functions described in the FSAR for Design Basis Accidents. Specifically, the design basis for the protective, safety related features of the plant are based on automatic action at the initial stages of Design Basis Accidents. This has been demonstrated in NEDO 10139, D.G. Scapini, et al, "Compliance of Protection Systems to Industry Criteria: GE BWR Nuclear Steam Supply System", June, 1970. For the Design Basis Accident conditions encountered as part of the scope of coverage for the new EOP's, the existing instrumentation was found to be acceptable as currently constituted.

Thus, NMPC's overall conclusion based on the above evaluations is that no variables should be considered Type A; and Category 1 criteria do not apply, even though the intent of Category 1 criteria was met for the cases considered. For degraded accident conditions beyond Design Basis Accidents, final review of instrumentation needs will be carried out in January/February, 1986, when the final EOP revisions are completed and the final step of the DCRDR is carried out, as described in the Introduction, above.

2. <u>Neutron Flux</u>

° NRC Contractor's Full Statement:

"Regulatory Guide 1.97 recommends environmentally qualified instrumentation. The licensee has instrumentation for this variable that has not been environmentally qualified. The licensee states that protective action is initiated prior to exposure to a harsh environment.

In the process of our review of neutron flux instrumentation for boiling water reactors (BWRs), we note that the mechanical drives of the detectors have not satisfied the environmental

qualification requirement of Regulatory Guide 1.97. A Category 1 system that meets all the criteria of Regulatory Guide 1.97 is an industry development item. Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The licensee should follow industry development of this equipment, evaluate newly developed equipment, and install Category 1 instrumentation when it becomes available."

- NRC Contractor's Summary Concern:

  "Neutron flux--the licensee's present instrumentation is acceptable on an interim basis until Category 1 instrumentation is developed and installed."

- NMPC Elaboration:

  Environmental qualification requirements, as detailed in 10CFR50.49, involve consideration of harsh environments beyond those encountered during normal operation. These can be created by postulated high energy line breaks or a LOCA. In this case, the NRC contractor's concern involves the SRM's and IRM's, where drive mechanisms are provided to move the detectors in and out of the core depending upon neutron flux levels. During normal full power operation, the detectors are withdrawn from the reactor. When a scram occurs, the detectors are immediately moved into the reactor in order to track power level decay and to verify shutdown. This process is initiated within seconds and recorder re-ranging/detector insertion continues over a period of a few minutes as neutron flux continues to decay. The drive equipment and connecting cable is located under the Reactor Pressure Vessel and would be subjected to the steam filling the containment for a brief period while detectors were being inserted. This is not likely to have any significant effects on the equipment during this time, particularly in the more likely event that the leak is small. This equipment is not fragile and has performed well in 15 years of plant operation with normal conditions inside the containment. While these conditions are not considered harsh by Environmental Qualification standards, they are certainly more severe than normal conditions outside the containment. Thus, the equipment should be considered to have some measure of qualification from its past operational history.

  It should also be recognized that there are several other independent indications of shutdown from the numerous rod position indication mechanisms, including annunciation of scram trip functions, etc. Furthermore, there are 4 SRM's and 8 IRM's, any one of which could provide shutdown corroboration. Finally, the APRM/LPRM's, which are not required to move at all, can monitor power down to 2-3%, which would adequately cover any significant postulated accidents.

  Relative to NMPC's Environmental Qualification program for NMP-1, the flux monitor drives were neither considered nor

covered because the formative evaluation carried out in scoping the program and determining which equipment to cover did not reveal any potential effects of significant concern. (For references, see item B. 5.)

3. Drywell Pressure

° NRC Contractor's Full Statement:

"Regulatory Guide 1.97 recommends monitoring the pressure in the drywell. The range recommended is from 12 psia to design pressure (62 psig). Category 1 instrumentation is recommended. The instrumentation supplied for this variable has a range of 0 to 75 psig, is not environmentally qualified and the redundant channels have a common power supply (thus the channels are not fully redundant).

The licensee does not justify not having instrumentation that covers from 12 psia to 0 psig. The licensee considers environmental qualification of the pressure transmitters unnecessary, as 0 to 250 psig transmitters, stated as being capable of serving the same function, are environmentally qualified. These instruments are described as being in the drywell for the variable primary containment pressure. The licensee considers the common power supply acceptable, as this instrumentation does not initiate automatic protective actions.

The licensee should either provide for monitoring of subatmospheric pressures or provide justification for not monitoring them.

Environmental qualification has been clarified since Revision 2 of Regulatory Guide 1.97 was issued. The clarification is in the environmental qualification rule, 10 CFR 50.49. It is concluded that the guidance of Regulatory Guide 1.97 has been superseded by a regulatory requirement. Any exception to this rule is beyond the scope of this review and should be addressed in accordance with 10 CFR 50.49.

We find the common power supply not acceptable because the primary containment pressure instrumentation (drywell pressure), which has redundant channels, does not cover the subatmospheric portion of the recommended range. The licensee should provide redundant power supplies for this variable."

° NRC Contractor's Summary Concern:

"Drywell pressure--the licensee should either provide for monitoring of subatmospheric pressures or justify not monitoring them; environmental qualification should be addressed in accordance with 10 CFR 50.49."

° NMPC Elaboration:

As shown in the Type C variable list in the April 2, 1984 submittal, NMP-1 has instrumentation labeled "Primary Containment Pressure" which for NMP-1 monitors the same thing as

"Drywell Pressure". This instrumentation meets all of the Category 1 criteria, including redundant safety related power supplies, except that subatmospheric monitoring capability was not identified in the table. However, a later change in the range of this instrumentation was made and it is now calibrated for -5 to +250 psig. This eliminates the identified deviation from Category 1 criteria for "Drywell Pressure".

Relative to Environmental Qualification, this instrumentation was included in NMP-1's program in response to 10 CFR 50.49 and is qualified. For references, see item B. 5.

4. Primary Containment Pressure

- NRC Contractor's Fuel Statement:

  "Regulatory Guide 1.97 recommends instrumentation for this variable with a range of from 10 psia to four times the design pressure of 62 psig (248 psig). The licensee has redundant instrumentation in the drywell with a range of 0 to 250 psig (the subatmospheric 10 psia to 0 psig is not measured) and non-redundant instrumentation in the torus with a range of 0 to 4 psig.

  The licensee has not provided justification for not monitoring any subatmospheric pressure. For the torus instrumentation, they state that instrumentation with a higher range is not necessary, even though the torus design pressure is 35 psig. This is because of vacuum breakers between the torus and the drywell that keeps the torus within 3 psi of the drywell. Thus, the drywell pressure instrumentation is applicable to the torus pressure. We conclude that with the exception of subatmospheric pressures, the instrumentation provided for this variable is acceptable. The licensee should either provide for the monitoring of subatmospheric pressures or provide justification for not monitoring them."

- NRC Contractor's Summary Concern:

  "Primary containment pressure--the licensee should either provide for monitoring of subatmospheric pressures or justify not monitoring them; redundant power supplies should be provided."

- NMPC Elaboration:

  As explained in the previous item, Primary Containment Pressure and Drywell Pressure are the same and the previous discussion adequately covers this deviation, also.

5. <u>Primary Containment Isolation Valve Position</u>

- NRC Contractor's Full Statement:

  "Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. Thus, environmental qualification, seismic qualification and redundancy are recommended for this instrumentation. The licensee provides instrumentation for these variables, however, deviations are identified in the above criteria.

  The licensee states that environmental qualification of those valves which are normally in their accident mitigation position is not needed as the valves are not required to change state during an accident.

  Environmental qualification has been clarified since Revision 2 of Regulatory Guide 1.97 was issued. The clarification is in the environmental qualification rule, 10 CFR 50.49. It is concluded that the guidance of Reuglatory Guide 1.97 has been superseded by a regulatory requirement. Any exception to this rule is beyond the scope of this review and should be addressed in accordance with 10 CFR 50.49.

  The licensee states that the seismic qualification of all isolation valve position switches has not been substantiated. The mounting of the position switches is seismically designed and installed on isolation valves and the licensee states that this provides assurance that the switches will remain operable following seismic activity. The isolation valve, its actuator, and its limit switches were typically procured as a unit with seismic specifications applied.

  We have no basis to believe that non-seismically qualified position switches will operate after a seismic event because they are installed on seismically qualified valves. Therefore, for those position swithes that are not seismically qualified to the original plant licensing requirements, the licensee should commit to replacing or upgrading the existing non-qualified components with seismically qualified parts.

  From the information provided, we find the applicant deviates from a strict interpretation of the Category 1 redundancy recommendation. Only the active valves have position indication (i.e., check valves have no position indication). Since redundant isolation valves are provided, we find that redundant indication per valve is not intended by the regulatory guide. Position indication of check valves is specifically excluded by Table 2 of Regulatory Guide 1.97. Therefore, we find that the redundancy for this variable is acceptable."

° NRC Contractor's Summary Concern:

Primary containment isolation valve position--environmental qualification should be addressed in accordance with 10 CFR 50.49; the licensee should upgrade non-seismically qualified position switches to include seismic qualification.

° NMPC Elaboration:

There are about 130 reactor & containment isolation valves that fall into this category, per the updated list proposed and submitted for the NMP-1 Technical Specifications on August 27 1984. This list was used as a basis for the Master List in the Environmental Qualification program. (i.e. This covered all equipment to be considered for EQ coverage.) Most of these were covered in the EQ program by being (or will be) replaced or otherwise shown to meet EQ requirements, as set forth in 10 CFR 50.49. The remaining 28 valves from the updated Technical Specification list were deleted from the EQ list on the basis of the evaluations made to determine which equipment needed to be included in the program. Program details, including these evaluations were reported in NMPC's submittals of May 20, 1983 and May 31, 1984; and they were accepted by the NRC per their SER of January 10, 1985.

Since the programmatic considerations involved were essentially the same, seismic considerations were also included in the Environmental Qualification program. Thus, NMPC was able to verify that reactor & containment isolation valve position switches meet seismic requirements except for the self actuated check valves which have no position switches and 28 others delineated in the previous paragraph. Of these 28 deleted from the EQ list, all except 3 are limitorque, solenoid, or small (1" diameter) air operated valves with positions switches built-in as an integral internal part of the valve itself. As stated in the April 2, 1984 submittal, the valves were originally purchased for the plant as one unit with the original plant seismic specification. NMPC is not able to corroborate that individual pieces of the valve mechanisms were individually qualified, but NMPC's judgement about the position switch part of the valve mechanisms is that there are no unusual weaknesses or potential problem areas that would show up during postulated design basis earthquake conditions. i.e., This is standard high quality nuclear industry equipment and it has performed well, to date. Based on the original plant design specifications attached to the Purchase Order for these valves, NMPC believes that the position switches are as well qualified, seismically, as the valves themselves.

The remaining three valves are the vacuum relief valves from the reactor building to the torus, which have micro position switches mounted on the outside of the valve body. The valves are normally closed and only operational under degraded conditions, not expected to be encountered during the DBA-LOCA event described in the FSAR. Furthermore, these valves are in series with another set of valves that are self actuating

checks. These check valves also have air operators and, thus, position switches; and these position switches have been both seismically & environmentally qualified, per NMP-1's EQ program. This provides adequate assurance for indicating containment isolation during postulated design basis earthquake conditions.

However, balanced judgements about systems/equipment operation during seismic events can only be made by considering all of the equipment operating together in such events. Thus, it is clear that consideration of the seismic capabilities of position switches should be folded in with the more comprehensive program covering affected equipment now being developed by the NRC to resolve USI A-46. Based on the recent NRC memorandum, T.P. Speis to A. R. Denton, "Implementation Plan for USI A-46", September 5, 1985, it appears that such a resolution program is about to be formally proposed for implementation. If need be, further consideration of this subject should be deferred to this implementation plan.

Meanwhile, the industry's thinking and approaches for resolving this concern have been well stated in NUREG/CP-0070 BNL-NUREG 51924, "Proceedings of the Workshop on Seismic and Dynamic Fragility of Nuclear Power Plant components", C. H. Hofmayer & K. K. Bandyopadbyay, Editors, August, 1985. NMPC has been following/participating in such activities and will consider them in their future program planning.

6. <u>Suppression·Pool·Water·Level</u>

° NRC Contractor's Full Statement:

"Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the bottom of the ECCS suction line to five feet above the normal water level. The licensee's instrumentation has a range from 3 ft. 3 in. below the ECCS suction to 3 ft. 8.5 in. above the normal water level. The licensee has not justified the deviation in the upper limit of the range.

The licensee has not shown that the provided range will not be exceeded. We conclude that the licensee should either re-range the instrumentation to include the recommended range, or provide justification for not doing so."

° NRC Contractor's Summary Concern:

"Suppression pool water level--the licensee should either re-range the existing instrumentation or provide justification for not doing so."

° NMPC Elaboration:

The torus is 27' in section diameter and has a 123' centerline circumference. Thus, the volume of the torus is very large compared to potential sources of water during postulated Design

Basis Accident conditions; and torus (quiescent) water level would not be expected to rise more than several inches for most scenarios. Even in extreme, degraded conditions where the entire Reactor Pressure Vessel inventory was assumed to end up in the torus along with some recirculation sytem and injected water, the torus level would not be expected to rise by more than 1 1/2'. On this basis, the upper range of the instrumentation at 3' 8 1/2" above the normal level is considered to be more than adequate for postulated highly degraded accidents. Furthermore, any remedial actions that the operator might take as a result of level being too high would be initiated long before the top end of the instrumentation range were reached. Considered another way, no further actions would need to be taken by the operator as a result of reaching the top end of the instrumentation range, all remedial actions having since been taken considerably before that time. It should also be noted that this instrumentation was previously upgraded in response to item II.F.1.5 of NUREG 0737, and that this upgrade was accepted by the NRC per their letter of July 18, 1983 to NMPC.

7. Radiation Exposure Rate

- NRC Contractor's Full Statement:

"Regulatory Guide 1.97, Revision 2, specifies Category 2 instrumentation for this variable with a range of 10^{-1} to 10^{4} R/hr. The licensee has provided instrumentation for this variable with ranges that vary, dependent on location, from the recommended range. The licensee has stated that containment breach is detected by the noble gas effluent monitors, and that release assessment is better performed with portable radiation instruments and secondary containment sample analysis. The licensee concludes that Category 3 instrumentation is adequate for the radiation exposure rate instrumentation.

Regulatory Guide 1.97, Revision 3 (Reference 5), changes this variable to Category 3. Therefore, the only deviation of the Nine Mile Point station for this variable is the range supplied for a given location. While supplying plant specific ranges, the licensee has not shown any analysis of radiation levels expected for the monitor locations.

The licensee should show that the existing radiation exposure rate monitors have ranges that encompass the expected radiation levels in their locations."

- NRC Contractor's Summary Concern:

"Radiation exposure rate--the licensee should show that the ranges supplied for this variable encompass the radiation level at the instrument location."

- NMPC Elaboration:

The number of existing radiation exposure rate monitors or area radiation monitors (ARMs) that are non-primary containment at

Nine Mile Point Unit 1 totals 33. They are listed in Table 1. These are manufactured by General Electric and are of G.M. Type with the following breakdown of ranges:

Range:	No. of Monitors	
0.01 - 100 mR/hr	3	
0.10 - 1000 mR/hr	29	
0.10 - 1000 mR/hr (Low)	1	(Monitor #17 located on Bridge Operators Platform in Rx Bldg. Elev. 340)
0.01 - 10,000 mR/hr (High)	···	
Total	33	

Twenty-one of these 33 monitors have ranges that would encompass the expected radiation levels in their locations. This determination is based upon a shielding study conducted by NES for Nine Mile Point 1 in 1980. The source terms and containment leakages assumed for this study were extremely conservative and represent highly degraded accident conditions for a total core failure. The results of this study have been incorporated into a site procedure to allow necessary operating activities to proceed in various areas of the facility during accident conditions.

Six of these 21 monitors are located within areas that would be designated "Unrestricted Access." i.e. Area dose rates are not anticipated to exceed 15 mR/hr in these areas and required periodic Health Physics surveys are sufficient to verify conditions there.

The remaining 15 of these 21 monitors would be in locations that could exceed 15 mR/hr and are areas that would be designated as "Restricted Access." i.e. Any area labeled as restricted access would not normally contain any large source of radiation. However, such areas have the possibility of becoming inaccessible through additional equipment failure, e.g., leakage at the main steam or feedwater isolation valves. Restricted areas are regarded as potentially containing significant amounts of radiation under degraded conditions but can probably be entered after being surveyed for appropriate protective measures.

These above mentioned 21 monitors (6 in "Unrestricted Access" areas and 15 in "Restricted Access" areas) all have ranges below the level of 10^4R as specified in R.G. 1.97, but have existing ranges that encompass the expected radiation levels in their locations. Therefore, they should meet the guidelines of R.G. 1.97.

The remaining 12 area radiation monitors are all located within the Reactor Building (secondary containment) and their various locations are shown in Table 2. With the possible exception of monitor #17 (high range), none of these monitors have ranges that would encompass the expected radiation levels in their locations; and therefore, may not meet the guidelines of R.G. 1.97. However:

1. The use of ARMs to detect a primary containment breach or leakage, per R.G. 1.97, is not only impractical but also unnecessary. This is because secondary containment radiation exposure rates would be more a function of radioactivity in the primary containment and in the liquids flowing through emergency system pipes, resulting in direct radiation shine on area monitors. Also, ARM's would give at the very most, ambiguous indications about potential containment leakage due to the widely scattered location of pipes and number of electrical penetrations.

2. High levels of airborne dose rates (beyond the current range) in the Reactor Building would:

   a) Preclude access per procedures
   b) Render the reliability of Reactor Building ARM indications suspect due to probable contamination of the detectors in the Reactor Building.

3. If airborne dose rates were low enough so as not to preclude access to the Reactor Building, such access would not likely be required to service safety-related equipment in a post accident situation. (i.e. Proper operation of safety related equipment will preclude generation of high level source terms.)

4. Since the Reactor Building would be designated as a Prohibited Access area, accessibility would be re-established by a combination of portable dose rate survey instruments and post accident sampling of the secondary containment atmosphere by Radiation Protection personnel. The existing ARMs (typically 4 decades lower than the R.G. 1.97 range) would be used only after radiation levels were within their range and their reliability had been re-established.

Thus, NMPC believes that the current instrumentation is adequate to handle anticipated operational needs for the expected dose rates under various accident conditions. It is also clear that containment leakage will be most clearly seen from the monitors in the plant stack. These monitors meet the guidelines of R.G. 1.97 as described in items 11 & 12 below.

Table 1

AREA RADIATION MONITOR DETECTOR LOCATIONS

Monitor No.	Location	Range of Monitor (mR/h)	Post LOCA Radiation: Zone	Expected Levels
1	Administration Building Entrance to Turbine Building	0.01 - 100	Unrestricted Access	< 15 mr/hr.
3	Reactor Control Room	0.01 - 100	"	"
4	Turbine Operating Floor Entrance (Generator End)	0.1 - 1000	Restricted Access	> 15 mr/hr.
5	Turbine Operating Floor Entrance (Feed Pump End)	0.1 - 1000	"	"
6	Condensate Pump Area (Valve Corridor)	0.1 - 1000	"	"
7	Feed Pump Area	0.1 - 1000	"	"
8	Electrical Switchgear Area - (Opposite Air Ejectors)	0.1 - 1000	"	"
9	Condensate Demineralizer Valve Area	0.1 - 1000	"	
10	Regeneration Area	0.1 - 1000	Unrestricted Access	< 15 mr/hr.
11	Makeup Demineralizer Area	0.1 - 1000	Restricted Access	> 15 mr/hr.
12	Waste Disposal (Loading Station Operational Areas, Convey Aisle)	0.1 - 1000	"	"
13	Waste Disposal - Pump Room	0.1 - 1000	"	"
14	Waste Disposal - Control Room	0.01 - 1000	"	"
15	Waste Disposal - Storage & Shipping Area	0.1 - 1000	"	"
24	Results Shop	0.1 - 1000	Unrestricted Access	< 15 mr/hr.
25	Decontamination Area-Large Equipment	0.1 - 1000	"	"
27	High Level Laboratory	0.1 - 1000	"	"
31	W. Bldg. Decontamination Sink	0.1 - 1000	Restricted Access	> 15 mr/hr.
32	W. Bldg. General Area - 247' El.	0.1 - 1000	"	"
33	W. Bldg. General Area - 229' El.	0.1 - 1000	"	"
34	Off Gas Bldg. Elevation 229'	0.1 - 1000	"	"

Table 2

REACTOR BUILDING AREA RADIATION MONITORS

Arm No.	Location	mRem/hr	R/hr	Post LOCA Radiation: Zone/Expected Levels
2	Reactor Building 318' The new fuel storage area South side of North Hall in Room	0.01 - 100	10^{-5} - 10^{-1}	Prohibited Access >1R/hr
16	Reactor Building 249' The TIP area - Inner TIP Room on West WAll	0.1 - 1000	10^{-4} - 1.0	"
17 (low)	Reactor Building 340' on Bridge-Operator's Platform-East Side of Bridge	0.1 - 1000	10^{-4} - 1.0	"
17 (high)	Reactor Building 340' On Bridge-Operator's Platform-East Side of Bridge	10 - 10^{6}	10^{-2} - 10^{3}	"
18	Reactor Building 340' on Emerg. Cond. Shield Wall at S.E. Corner of Equipment Hatch	0.1 - 1000	10^{-4} - 1.0	"
19	Reactor Building 198' N.E. The reactor building equipment drain tank area on North wall	0.1 - 1000	10^{-4} - 1.0	"
20	Reactor Building 298' West The reactor building closed-loop cooling area - On column between RCLC heat exchangers	0.1 - 1000	10^{-4} - 1.0	"
21	Reactor Building 261' N.D. The reactor cleanup system area on north wall by CU pumps	0.1 - 1000	10^{-4} - 1.0	"
22	Reactor Building 281' NE The reactor fuel pool cooling system area on column in N.E. area	0.1 - 1000	10^{-4} - 1.0	"
23	Reactor Building 237' N.W. Control rod drive module area - On west wall near N.W. stairs	0.1 - 1000	10^{-4} - 1.0	"
26	Reactor Building 340' East The spent fuel pool area - east wall	0.1 - 1000	10^{-4} - 1.0	"
28	Reactor Building 318' N.W. The containment spray heat exchanger	0.1 - 1000	10^{-4} - 1.0	"
29	Reactor Building 237' The reactor north instrumentation room on south wall	0.1 - 1000	10^{-4} - 1.0	"

8. Drywell Atmospheric Temperature

- NRC Contractor's Full Statement:

"Regulatory 1.97 recommends instrumentation for this variable with a range of 40 to 440°F. The licensee has instrumentation for this variable with a range of 50 to 300°F, and states that the range is sufficient to provide the operator with information relative to the potential for flashing in the level sensing instrument lines.

We agree that the given range is sufficient to monitor the potential for flashing in the instrumentation lines for reactor vessel level.

Our examination of the Final Safety Analysis Report (FSAR, Reference 6) shows that the maximum internal drywell design temperature is 310°F. The actual peak temperature would be less than this and of short duration. Based on this, the licensee's upper limit of 300°F for the post accident period is sufficient. We have no basis on which to accept the lower limit of 50°F rather than the recommended 40°F.

We conclude that the licensee should justify this deviation from the recommended range or re-span the instrumentation to coincide with the range recommended by Regulatory Guide 1.97.

- NRC Contractor's Summary Concern:

"Drywell atmospheric temperature--the licensee should justify a deviation from the recommended range or supply the recommended range."

- NMPC Elaboration:

Since the reactor was started up in 1969, the drywell atmosphere has not dropped below 50°F and there are no conditions under which NMPC can envision where this would occur during accident conditions or during shutdown with all access openings actuated. The sources of heat are too great for this to occur given the relatively small volume of the drywell.

9. & Residual Heat Removal System Flow

10. Residual Heat Removal Heat Exchanger Outlet Temperature

- NRC Contractor's Full Statement:

"Regulatory Guide 1.97 recommends monitoring the residual heat removal system for flow (0 to 110 percent of design flow) and heat exchanger outlet temperature (32 to 350°F) with environmentally qualified instrumentation. Unit 1 at Nine Mile Point has no direct indication of flow rate for this variable. The licensee states that the shutdown cooling system flow is manually adjusted to maintain the cooldown rate below 100°F/hr.

Thus, flow is controlled by the shutdown cooling system temperature. Loss of flow is not indicated in this manner. Therefore, we find that the licensee should provide the recommended flow instrumentation.

Individual heat exchanger outlet temperatures have ranges of 40 to 400°F, the common header temperature instrumentation has a range of 0 to 400°F. Thus the recommended temperature range is satisfied.

The instrumentation is not environmentally qualified. The licensee states that the system does not mitigate the consequences of a loss of coolant accident or a high energy line break. Environmental qualification has been clarified since Revision 2 of Regulatory Guide 1.97 was issued. The clarification is in the environmental qualification rule, 10CFR50.49. It is concluded that the guidance of Regulatory Guide 1.97 has been superseded by a regulatory requirement. Any exception to this rule is beyond the scope of this review and should be addressed in accordance with 10CFR50.49."

- NRC Contractor's Summary Concern:

  "Shutdown cooling system flow--the licensee should provide the recommended instrumentation."

  "Shutdown cooling system temperature--environmental qualification of this instrumentation should be addressed in accordance with 10CFR50.49."

- NMPC Elaboration:

  First, it should be noted that Nine Mile Point 1 has a Shutdown Cooling System (SCS), not a Residual Heat Removal System. This system is essentially a small scale parallel recirculation loop with the functional exception of an in-line heat exchanger to remove heat. It has none of the extra appurtenances and functions of a Residual Heat Removal system and thus, is not as subject to potentially conflicting interactions or accidents that might disturb flow in the system. i.e. It is a simpler, more direct system.

  Second, there is considerable instrumentation and redundancy in the system as indicated in the attached system diagram from Chapter X of the FSAR. (Fig. 1)

  Flow disturbances can occur from a number of initiating events such as pipe leaks, pump seizures, and blockages of various kinds. Considering the range of possibilities, there does not appear to be any credible mechanism where flow disturbances would not be seen with existing instrumentation. This includes SCS inventory losses causing an unexplained drop in reactor water level. This would be seen on reactor water level monitors which are not shown in the diagram. Temperature elements are positioned at several locations on the system and could indicate

a variety of possible inventory losses or stoppages. SCS pump operating information is also available to provide indications of other kinds of potential flow anomolies (pump seizures, etc.). There are also indirect indications available from the instrumentation on the Reactor Building Closed Loop Cooling Water System. The Reactor Building Closed Loop Cooling Water System is particularly sensitive to flow/temperature changes on the tube side since the shell side cooling that it provides for the SCS heat exchanger is its major load

Finally, the SCS would not be expected to be operating during postulated accident conditions. During cooldown from an accident, extended indefinitely, if need be, heat can be removed from the reactor by blowdown (or drainage) into the torus, and heat can be removed from the torus through redundant containment spray heat exchangers. In turn, these heat exchangers are cooled by separate redundant containment spray raw water pumps at the plant intake canal which are powered by the emergency Diesel Generators. Ample indications of the operation of this safety related equipment is available in the Control Room. An intertie can also be opened to pump raw water directly into the core and containment spray system in extremely degraded conditions.

In summary, NMPC's evaluations did not reveal any credible, significant flow disturbances in the SCS that could not be seen by existing instrumentation. Further, any such situation would not be expected to occur until after the plant was in a normal stable shutdown cooling situation, not associated with an accident condition, and easily seen and handled within time spans not immediately threatening to the reactor.

Relative to Environmental Qualification, the NRC specifically excluded consideration of cold shutdown equipment (which is the category that the Shutdown Cooling System falls into) from 10CFR50.49 as described in Federal Register, Vol. 48, No. 15, page 2731. In essence, this states that concerns about the adequacy and reliability of shutdown decay heat removal systems will be addressed by the resolution to USI A-45 and need not be included in the EQ rule. Thus, Environmental Qualification has been adequately addressed by a superseding regulatory action.

11. Noble Gas and Vent Flow Rate--Common Plant Vent

- NRC Contractor's Full Statement:

  "Regulatory Guide 1.97 recommends instrumentation for this variable with a range from 10^{-6} to 10^{+4} uCi/cc. The licensee has provided instrumentation with a range that goes up to 10^{+3} uCi/cc.

  The licensee indicates that their evaluation of the capabilities of this system is not complete and that changes regarding the range of the equipment may occur. They further state that several other monitors are available to monitor stack releases

with ranges from 10^{-1} to 10^{6} counts per second and 10^{-1} to 10^{6} counts per minute. They did not indicate that the alternate instrumentation will register up to 10^{+4} uCi/cc. Therefore, we cannot accept this deviation. The licensee should report on the final configuration, describing modifications to bring the range into compliance with the regulatory guide or provide additional justification for not doing so."

- NRC Contractor's Summary Concern:

  "Noble gas and vent flow rate-common plant vent--the licensee should provide the recommended range or provide justification for not doing so."

- NMPC Elaboration:
  Combined with 12. below.

12. <u>Particulates and Halogens--All Identified Plant Release Points</u>

- NRC Contractor's Full Statement:

  "Regulatory Guide 1.07 recommends instrumentation for this variable with a range from 10^{-3} to 10^{+2} uCi/cc. The licensee has provided instrumentation for this variable with a range from 10^{-3} to 10 uCi/cc. They have not supplied justification for the deviation from the recommended upper limit of the range.

  We conclude that the licensee should either provide instrumentation that covers up to the recommended upper limit of the range or provide justification for accepting the present instrumentation."

- NRC Contractor's Summary Concern:

  "Particulates and halogens--all identified plant release points--the licensee should provide instrumentation of the recommended range or justification for the deviation in the upper limit of the range."

- NMPC Elaboration (includes item 11. above):

  Nine Mile Point 1 installed a new gaseous effluent monitoring system (SAIC's Model 400 Radioactive Gaseous Effluent Monitoring System) in 1983 which automatically samples and isotopically analyzes stack particulate, iodines and noble gases. This system is capable of remotely analyzing diluted or undiluted samples with activities ranging from 10^{-13} to 10^{+1}, 10^{-13} to 10^{+1}, and 2×10^{-8} to 10^{+5} uCi/cc for particulates, iodines and noble gases, respectively. Sample flow can be maintained isokinetic in the range of approximately 15% - 110% vent (stack) flow. Below 15% vent flow, samples can still be analyzed under non-isokinetic conditions.

In the event iodine (or particulate) effluent activities exceed 10^{+1} uCi/cc, samples can still be acquired using the gaseous effluent monitoring system. Sample analysis can be accomplished by removing the high activity sample from the gaseous effluent monitoring system using remote handling tools (or other means consistent with ALARA) and transporting the sample to the laboratory for high activity isotopic analysis. Sample activities as high as approximately $2 \times 10^{+3}$ uCi/cc can be analyzed in the lab using gamma spectroscopy with the sample source positioned 100 cm from the detector crystal.

13. Plant·and·Environs·Radioactivity

° NRC Contractor's Full Statement:

"Revision 2 of Regulatory Guide 1.97 recommends a multichannel gamma-ray spectrometer for this variable for isotopic analysis in release assessment and analysis.

The licensee has not provided the information required by Section 6.2 of NUREG-0737, Supplement No. 1. for this variable. Their equipment evaluation was in progress when Reference 4 was submitted. We conclude that the licensee should provide the information required, identify any deviation from the regulatory guide recommendations and provide satisfactory justification for any deviation."

° NRC Contractor's Summary Concern:

"Plant and environs radioactivity--the licensee should provide the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviation from the recommendations of Regulatory Guide 1.97, and provide satisfactory justification for any deviation."

° NMPC Elaboration:

Niagara Mohawk has recently purchased a portable gamma-ray spectroscopy system from Canberra Ind. Inc. The major components of this system are:

°An 8192 channel MCA capable of performing simple user defined subroutines,

°A 2" by 2" NaI detector,

°A 15% relative efficiency high purity germanium detector,

°A cassette recorder for storing spectral data in the field, and

°A second cassette recorder for transferring field data into the memory of a Canberra series 90 MCA associated with a whole body counter. The entire spectroscopy system, excluding the high purity germanium detector, will be housed in a heavy duty aluminum carrying case. The spectroscopy system is capable of

operating using a rechargeable battery pack, standard AC power and a car battery. Also provided as part of this system, will be a 2" thick lead counting well and an adjustable tripod for ground deposition measurements. This meets the guidelines for Category 3 instrumenation, per R.G. 1.97.

C. Verification

As noted in the cover letter to NMPC's April 2, 1984 submittal, internal verification checks had not been completed at the time of the submittal. This was later completed and some clarification of the details contained in the tables was noted. None of these affect the evaluations or conclusions. However, for completeness, these notes are listed below.

Variable	Clarification
Neutron Flux	°SRM/IRM instrumentation is powered from the 24VDC system, which is supplied from batteries and battery chargers backed-up by the RPS 11&12 power supplies. The 24 VDC system is also class IE. °The drives themselves are powered by AC Power Board 167, which is also class IE. °There are also flux monitor meters on a back (G) panel in the Control Room.
Coolant Level in Reactor	°Under Redundancy & Sensor Location, add the words "starting with" in front of the statement that begins with "Two level transmitters..."
Primary Containment Isolation Valve Position Switches	°The isolation valve indicating lights on the containment mimic on F panel are powered from the 28 VAC mimic bus which is powered through a transformer from RPS bus 11. °Many of the AC powered motor operated isolation valves have individual position indication lights in the Control Room (separate from the mimic lights) powered by the associated line voltage. These are class IE power supplies.
Primary Containment Pressure	°This instrumentation has been re-ranged to -5 to +250 psig as discussed in item B.3.