UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, *ET AL.*	)	Docket No. 50-289 (Restart)
	)	
(Three Mile Island Nuclear Station, Unit 1)	)	

NRC STAFF'S RESPONSES TO UCS
INTERROGATORIES OF SEPTEMBER 25, 1980 TO NRC STAFF

Attached are the NRC Staff's (Staff) responses to UCS Interrogatories of September 25, 1980 to the NRC Staff and the affidavit of those persons who prepared the responses.

Respectfully submitted,

James R. Tourtellotte
James R. Tourtellotte
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 14th day of November 1980

UCS Interrogatory #1

Item 2.1.1 of NUREG-0578 has been "clarified" in the September 5, 1980 version of Item II.E.3.1 of NUREG-0660 with a new requirement that redundant heater capacity must be provided.

a. What is the basis for this new requirement?

b. Does the Staff now take the position that functioning of the pressurizer heaters from the onsite power supply is important to safety?

c. If the answer to b. is no, explain why redundancy is required for components that are unimportant to safety and specify the Commission's regulation(s) that require(s) redundancy of non-safety related components.

d. If the answer to b. is now, give examples of other instances where redundancy was required for components classified as non-safety related and explain the basis for or purpose of each such requirement.

NRC Response to Interrogatory #1 (RGF)

Intervenor is correct in that the September 5, 1980 staff letter does not identify the requirement to provide redundant pressurizer heater capacity as a new requirement. However, this letter was a draft and was incorrect in so stating. The current version of the Action Plan Clarification dated October 31, 1980, NUREG-0737, contains a correct statement of position as discussed below.

The original requirement with respect to this aspect of the subject is found in NUREG-0578 Appendix A page A-4 paragraph 3.1.1 which states in part:

"...The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability."

This requirement was clarified in the October 30, 1979 staff letter providing clarification on all of the Lessons Learned requirements (NUREG-0578). Specifically, Section 2.1.1 Clarification #1 which states:

"In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class 1E division power supply."

As can be seen from comparing the October 30, 1979 with the October 31, 1980 version of the Clarification, the stated basis for requiring redundant heater capacity is not repeated in the October 31 version. The second half of each clarification remains identical. It is clear that both versions contain the same requirements.

It is therefore concluded that it is not necessary to further address the interrogatory and its various subparts based upon the demonstration above that the basis of the interrogatory was an acknowledged editorial error in a draft document.

UCS Interrogatory #2

Item 2.1.2 of NUREG-0578 has been "clarified" in the September 5, 1980 version of Item II.D.1 of NUREG-0660 with a new requirement for testing of the PORV block valve.

a. What is the basis for this new requirement?

b. Does the Staff take the position that isolation of a stuck open or leaking PORV is a function that is important to safety? Explain the reasons for your answer.

c. If the answer to b. is yes, explain why redundant block or isolation valves, classified a safety grade and automatically closed, are not required.

d. If the answer to b. is no, explain why testing of the PORV block valve is required and specify the Commission's regulation(s) that require(s) such testing of non-safety grade components.

NRC Response to Interrogatory #2 (EGH)

a. Block valves must be qualified to ensure that a stuck open relief valve can be isolated, thereby terminating a small loss of coolant accident due to a stuck open relief valve.

b. Isolation of a stuck open relief valve is not required to ensure safe plant shutdown. However, isolation capability under all fluid conditions that could be experienced under operating and accident conditions will result in a reduction in the number of challenges to the emergency core cooling system. Repeated and unnecessary challenges to these systems is undesirable.

c. Not applicable.

d. Testing of the PORV block valves is required for the reasons stated above and to ensure that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage as required by General Design Criteria 14.

UCS Interrogatory #3

Item 2.1.3.a of NUREG-0578 has been "clarified" in the September 5, 1980 version of Item II.D.3 of NUREG-0660 by noting that the relief and safety valve position indication should be seismically and environmentally qualified. In contrast, the TMI-1 Restart Evaluation Report (NUREG-0680) notes that, if the seismic and environmental qualification requirements will not be met by January 1, 1980 (sic), a proposed qualification schedule should be provided. (Page C8-12).

a. Explain the reasons for the difference between the "clarification" in the September 5, 1980 letter and the TMI-1 Restart Evaluation.

b. If the response to a. includes a reference to backup methods for indirectly determining relief or safety valve position, describe the extent to which that backup equipment meets safety grade requirements, including seismic and environmental qualification requirements. (Please state explicitly whether the response applies to the position indication for the relief valve, safety valve, or both.)

NRC Response to Interrogatory #3 (RGF)

There appears to be some confusion on the part of the Intervenor regarding this subject. The quote of NUREG-0680 in the interrogatory is in actuality a direct quote taken from the October 30, 1979 clarification letter which followed NUREG-0578. The September 5, 1980 letter simply moved the January 1, 1980 due date for operating reactors from the CLARIFICATION section to the APPLICABILITY section.

Part b. of the Interrogatory is already addressed in the TMI-1 Restart Evaluation Report (NUREG-0680).

UCS Interrogatory #4

The part of Item 2.1.3.b of NUREG-0578 which addresses new instrumentation for indication of inadequate core cooling has been "clarified" in the September 5, 1980 version of Item II.F.2 of NUREG-0660 with a new requirement. The new clarification Item No. 7 states that all instrumentation in the final inadequate core cooling monitoring system must be evaluated for conformance to Regulatory Guide 1.97, Revision 2. In addition, Clarification Item No. 6 has been changed.

a. Explain the reasons for the difference between Clarification Item No. 6 on page C8-20 of the TMI-1 Restart Evaluation and that in the September 5, 1980 version of Item II.F.2 of NUREG-0660.

b. Does the Staff propose to apply Clarification Item No. 7 of the September 5, 1980 version of Item II.F.2 of NUREG-0660 to TMI-1? If not, why not?

c. Provide a copy of Regulatory Guide 1.97, Revision 2 (or the most recent draft of it), a copy of all documents exchanged with the nuclear industry at the September 25-26, 1980 meeting held in Colorado to discuss Revision 2, and a copy of the Staff's minutes and/or summary of that meeting.

NRC Response to Interrogatory #4 (LEP)

The UCS Interrogatory #4 is based on the preliminary NRC "Clarification Letter" of September 5, 1980. This has since been modified by our letter to licensees and applicants from D. G. Eisenhut dated October 31, 1980, NUREG-0737. Our response is directed to the October 31 clarification letter, Item II.F.2 and II.F.2, Attachment 1, and Appendix A, copies of which are available in all NRC Public Document Rooms.

Changes to the previous requirements and guidance as stated in Item 2.1.3.b of NUREG-0578 and the clarification letter of October 30, 1979 are discussed in the October 31, 1980 clarification letter. Pertinent portions of Regulatory Guide 1.97 have been extracted and included as Appendix A to the clarification letter in order to avoid reference to a document which is not yet issued in final form. In addition to the change to Item 6 and the new clarification Item 7, new clarification Items 8, 9 and 10 and Attachment 1, which details the design criteria for PWR in-core thermocouples, have been added. An elaboration on the reason for these changes in response to the UCS interrogatory follows.

a. Item 6 - This item was expanded to make clear that the requirement for full range indication was not intended to exclude the use of diverse measurement methods for different portions of the range. There is no change in principle from the original Clarification Item No. 6 on page C8-20 of the TMI Restart Evaluation.

   Item 7 - Item 7 on page C8-17 of NUREG-0680 (TMI-1 Restart) indicated that the instrumentation (Saturation Meter) qualifications must meet the requirements of Regulatory Guide 1.97 in the long term. Item 7 was added to II.F.2 to make clear that Regulatory Guide 1.97 (in the form of Appendix A) is applicable to all instrumentation in the final inadequate core cooling (ICC) system.

   Item 8 - Specific requirements were developed for liquid level displays and associated hardware at locations available for maintenance. The purpose of this requirement was to make feasible the use of computer associated displays in a manner that would maintain liquid level indication reliability while facilitating the procurement and installation of such systems to meet the required implementation schedule.

   Item 9 - Specific design requirements were developed for PWR in-core thermocouples and were included as Attachment 1 to the final clarification letter.

   Item 10 - Human-factors design considerations for ICC displays and alarms were added to the clarification.

b. The staff is requiring that the total ICC instrumentation system be evaluated for conformance to the October 31, 1980 letter in a licensee submittal which is due on January 1, 1981. The staff will review that submittal to determine the acceptability of proposed implementation dates consistent with the schedular requirements of the clarification letter, including the "Applicability" section of Appendix A.

c. A copy of the current draft version of Regulatory Guide 1.97, and a summary (minutes) of the September 25-26, 1980 meeting in Colorado are enclosed.

UCS Interrogatory #5

Item 2.1.4 of NUREG-0578 has been "clarified" in the September 5, 1980 version of Item II.E.4.2 of NUREG-0660 with two new requirements (Positions 5 and 6) and substantive changes to the clarification items.

a. What are the bases for the new requirements set forth in Positions 5 and 6 of the September 5, 1980 version of Item II.E.4.2 of NUREG-0660?

b. Discuss the reasons for the changes in the September 5, 1980 Clarification Items compared to those set forth on page C8-22 of the TMI-1 Restart Evaluation.

c. Provide a copy of Regulatory Guide 1.141, Revision 1 or, if it has not been issued, the most recent draft made available to the nuclear industry.

d. Provide a copy of the additional guidance on the classification of essential vs. non-essential systems mentioned in the new Clarification Item No. 4.

NRC Response to Interrogatory #5 (MBF)

a. Positions 5, 6 and 7 of Item II.E.4.2 in the September 5, 1980 letter and in NUREG-0737 are not new requirements since they were included in the NUREG-0660 requirements. Those positions were added to the NUREG-0578 requirements (Positions 1 through 4 of II.E.4.2 in NUREG-0737) to provide additional assurance that the containment isolation system would function properly. The purpose of Position 5 is to increase the likelihood of isolating containment in even small pressure rises inside containment. Position 6 was added because recent operating experience has indicated that normally closed valves could be inadvertently left open. Implementation of this position would make more remote any inadvertent opening of these valves. Position 7, with which TMI-1 already complies, will provide additional assurance that contaminated air will not escape through the purge lines.

b. The items on page C8-22 of the TMI Restart Evaluation (NUREG-0680) are the short-term recommendations of NUREG-0578 (and are included in NUREG-0737 as positions 1 through 4 of Item II.E.4.2). The additional items in NUREG-0737 (Positions 5, 6 and 7) are not considered to be as important to safety as the short-term recommendations but they do provide an extra margin of safety that the staff feels is necessary for the long-term operation of nuclear power plants.

c. Regulatory Guide 1.141, Revision 1 was referenced in the September 5, 1980 letter as providing guidance on future isolation signal diversity requirements. NUREG-0737, which supercedes the September 5, 1980 letter as the implementation document on the TMI Action Plan, does not reference this guide because it was determined that the subject of the guide was outside the scope of the TMI Action Plan. Nevertheless, a copy of the latest version of this regulatory guide is attached for your use.

d. The discussion on additional guidance on the classification of essential vs. non-essential systems (Clarification 3 to Item II.E.4.2) in the September 5, 1980 letter has been modified in the final version of the TMI Action Plan (NUREG-0737). In NUREG-0737, we state in clarification 3 to II.E.4.2, that Regulatory Guide 1.141, Revision 2 will contain this guidance. This guide will be formulated next year.

UCS Interrogatory #6a

Items 2.1.7.a and 2.1.7.b of NUREG-0578 have been "clarified" in the September 5, 1980 version of Item II.E.1.2 of NUREG-0660 by specifying those requirements of IEEE Standard 279-1971 which must be met to comply with the position that automatic initiation of AFW and AFW flow indication must meet safety grade requirements.

a. For each requirement of Section 4 of IEEE Standard 279-1971 which is not listed in the Clarification section of the September 5, 1980 version of Item II.E.1.2 of NUREG-0660, explain the basis for not listing the requirement. (Please answer separately for the automatic initiation function and the flow indication or state that the answer applies to both.)

NRC Response to Interrogatory #6a (DFT)

The October 31, 1980 version of the Post TMI-requirements for Item II.E.1.2 of NUREG-0660; Part 1 (Auxiliary Feedwater System Automatic Initiation) states that the intent of the recommendation is to assure a reliable automatic initiation system for the auxiliary feedwater system. It is the staff's belief that this objective can be achieved by providing a system which meets all the requirements of IEEE Standard 279-1971. The paragraphs that are highlighted in the clarification are, as stated, the ones which are to be addressed, as a minimum, in the submittal of information required for staff review.

In addition, the staff believes that conformance to these highlighted paragraphs provides a basis to reach a reasonable assurance finding that the system is acceptable.

The paragraphs that were not highlighted are believed to be either:

1. Covered sufficiently by other highlighted paragraphs. In this category is paragraph 4.5 which is believed to be covered by paragraphs 4.3 and 4.4; and paragraph 4.8 which is believed to be covered by paragraph 4.1; or

2. Not of specific significance or not applicable to the AFW initiation system. In this category are paragraphs 4.14, 4.18, 4.19, 4.20, 4.21, and 4.22 which are not believed to be of specific significance to the AFW initiation system. Paragraph 4.15 is not believed to be applicable to the AFW system because there are no multiple setpoints. Paragraph 4.16 is not believed to be applicable to the AFW system because the system for the most part is manually controlled and once the system is actuated, operator normally takes control of the system.

The October 31, 1980 version of the Post TMI requirements for Item II.E.1.2 of NUREG-0660; Part 2 (Auxiliary Feedwater System Flowrate Indication) states that the intent of the recommendation was to assure a reliable indication of auxiliary feedwater system performance. In order to meet this objective, the clarification cites specific design principles which should be met by the flow indication system and the same is addressed in the submittal of information required for staff review. Included, for some specific cases, are certain requirements from IEEE 279-1971. The staff believes that conformance with the listed design principles can help achieve the stated objective. Also, these design principles have typically been applied to post-accident monitoring instruments (R.G. 1.97). All requirements of IEEE Standard 279 are not listed for the Flow Indication System because this system does not perform an automatic protection function and is, therefore, not strictly considered part of the protection system as defined in IEEE 279-1971.

UCS Interrogatory #6b

Is it the Staff's position that AFW flow indication is part of the protection system as defined in IEEE Standard 279-1971? If so, please explain fully the bases for that position. If not, explain the reasons for applying protection system requirements to equipment that is not part of the protection system.

NRC Response to Interrogatory #6b (DFT)

As stated in response to Interrogatory #6a, the staff does not consider the AFW flow indication system to be part of the protection system. The staff believes that conformance to the cited design principles, which have typically been applied to post-accident monitoring instruments, will help achieve a reliable indication system.

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

1st PMD 4/14/80

REGULATORY GUIDE 1.141

DRAFT

POOR ORIGINAL

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

A. INTRODUCTION

General Design Criteria 54, 55, 56, and 57 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require that piping systems penetrating primary reactor containment be provided with isolation capabilities that reflect the importance to safety of isolating these piping systems. This guide describes a method acceptable to the NRC staff for complying with the Commission's requirements with respect to containment isolation of fluid systems. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

Working Group ANS-56.2 of the American Nuclear Society Standards Committee ANS-50, Nuclear Power Plant Systems Engineering, has prepared a standard that specifies the minimum design requirements for containment isolation of fluid systems that penetrate the primary containment boundary of light-water-cooled reactors. This standard was approved by the American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, and designated ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems."[1]

The provisions of ANSI N271-1976 include minimum design, testing, and maintenance requirements for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors. Requirements for the design and testing of power supplies, qualifying of Class 1E equipment, and the design and testing of protection systems are outside the scope of this standard. These areas are not completely covered by the references given in ANSI N271-1976.

This standard contains requirements indicated by the verb "shall" and recommendations indicated by the verb "should." The recommendations as well as the requirements of the standard were evaluated with respect to importance to safety. All recommendations are considered to be of sufficient importance to safety to be endorsed along with the requirements given in the standard.

This revised guide includes improved regulatory guidance as a result of NRC staff review of the lessons learned from the Three Mile Island Nuclear Station Unit 2 accident. In particular, the review revealed that an isolation signal derived from containment pressure was not sufficient to ensure containment isolation when necessary. Radiation level within containment is the primary concern in protection of the public health and safety and should be monitored. In addition, this may be the only parameter capable of initiating containment isolation during certain situations (e.g., refueling operations). An isolation signal derived from actuation of an engineered-safety-feature system or subsystem is a reliable backup to ensure containment isolation under those conditions that warrant an engineered-safety-feature actuation. These three parameters (containment pressure, radiation level, and engineered-safety-feature actuation) provide diversity for containment isolation so as to prevent the release of radioactivity beyond the accepted limits under abnormal occurrences or credible accident conditions.

The manner in which the NRC staff will implement this regulatory guide is discussed in Section D, Implementation. In an effort to provide concise implementation guidance, Section D has been written in two parts. The first part addresses the implementation of regulatory positions 3, 4, and 5, which relate to the recommendations presented in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."[2] The imple-

*Lines indicate substantive changes from previous issue.

[1] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

[2] NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," published in July 1979, is available from the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or the National Technical Information Service, Springfield, Virginia 22161.

Modified Draft 3
~~October 30, 1980~~ ~~October 8, 1980~~
Nobember 6, 1980 Division 1
Task RS 917-4

Contact: A. S. Hintze, (301) 443-5913

[~~PROPOSED~~] 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

A. INTRODUCTION

Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," includes a requirement that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50 includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents, and that equipment, including the necessary instrumentation, at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor.

Criterion 64, "Monitoring Radioactivity Releases," of Appendix A to 10 CFR Part 50 includes a requirement that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant.

B. DISCUSSION

Indications of plant variables are required by the control room operating personnel during accident situations to (1) provide information required to permit the operator to take preplanned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, engineered-safety-feature systems, and manually initiated safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity); and (3) provide information to the operator that will enable him to determine the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) and if a gross breach of a barrier has occurred. In addition to the above, indications of plant variables which provide information on operation of plant safety systems and other systems important to safety are required by the control room operating personnel during an accident to (1) furnish data regarding the operation of plant systems in order that the operator can make appropriate decisions as to their use; and (2) provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

At the start of an accident, it may be difficult for the operator to determine immediately what accident has occurred or is occurring and, therefore, to determine the appropriate response. For this reason, reactor trip and certain other safety actions (e.g., emergency core cooling actuation, containment isolation, or depressurization) have been designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant variables required to enable the operation of manually initiated safety systems and other appropriate operator actions involving systems important to safety.

[~~Instrumentation-is-also-needed-to-provide-information-about-some-plant-parameters-that-is-currently-not-available-using-present-technology-will-alert-the-operator-to-conditions-that-have-degraded-beyond-those-postulated-in-the-accident-analysis.--In-particular,-it-is-important-that-the-operator-be-informed~~

~~regarding-that-status-of-coolant-level-in-the-reactor-vessel-or-the-existence of-core-voiding-thus-providing-indication-of-potential-degraded-core-cooling and-imminent-fuel-damage.--Direct-indication-of-coolant-level-in-the-reactor vessel-is-not-currently-available-in-pressurized-water-reactors.--However,-it-is imperative-that-this-capability-be-developed-within-a-reasonable-time-in-order to-provide-the-operator-with-this-vital-information-in-a-positive,-unambiguous manner.~~]

Independent of the above tasks, it is important that the operator be informed if the barriers to radioactive materials release are being challenged. Therefore, it is essential that instrument ranges be selected such that the instrument will always be on-scale. Narrow-range instruments may not have the necessary range to track the course of the accident, consequently, multiple instruments with overlapping ranges may be necessary. (In the past, some instrument ranges have been selected based on the set-point value for automatic protection or alarms.) It is essential that degraded conditions and their magnitude be identified so that the operator can take actions that are available to mitigate the consequences. It is not intended that the operator be encouraged to prematurely circumvent systems important to safety but that he be adequately informed in order that unplanned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade are loss-of-coolant accidents (LOCAs), overpressure transients, anticipated operational occurrences which become accidents such as anticipated transients without scram (ATWS), reactivity excursions which result in releases of radioactive materials. Such events require that the operator understand, within a short time period, the ability of the barriers to limit radioactivity release, i.e., the potential for breach of a barrier, or an actual breach of a barrier by an accident in progress.

It is essential that the required instrumentation be capable of surviving the accident environment in which it is located for the length of time its function is required. It could therefore either be designed to withstand the accident environment or be protected by a local protected environment.

It is important that accident-monitoring instrumentation components and their mounts that cannot be located in Seismic Category I buildings be designed to continue to function, to the extent feasible, following ~~during~~ seismic events. Consequently, it it is essential that they be designed to resist the effects of seismic excitation. An acceptable method for demonstrating the adequacy of the seismic resistance of this instrumentation would be to design ~~qualify~~ it to meet the seismic criteria applicable to like instrumentation installed in seismically qualified ~~at other~~ locations ~~in the plant.~~

Variables selected for accident monitoring can be selected to provide the essential information needed by the operator to determine if the plant safety functions are being performed. It is essential that the range selections be sufficiently great that the instruments will always be on scale. Further, it is prudent that a limited number of those variables which are functionally significant (e.g., containment pressure, primary system pressure) be monitored by instruments qualified to more stringent environmental requirements and with ranges that extend well beyond that which the selected variables can attain under limiting conditions; for example, a range for the containment pressure monitor extending to the burst pressure of the containment in order that the operator will not be unaware as to the pressure inside containment. Provisions of such instruments are important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions determined. It is also necessary to be sure that when a range is extended, the sensitivity and accuracy of the instrument are within acceptable limits for monitoring the extended range.

Normal power plant instrumentation remaining functional for all accident conditions can provide indication, records, and (with certain types of instruments) time-history responses for many variables important to following the course of the accident. Therefore, it is prudent to select the required accident-monitoring instrumentation from the normal power plant instrumentation to enable the operator to use, during accident situations, instruments with which he is most familiar. Since some accidents could impose severe operating requirements on instrumentation components, it may be necessary to upgrade those normal power plant instrumentation components to withstand the more severe operating conditions and to measure greater variations of monitored variables that may be associated with an accident. It is essential that instrumentation so upgraded does not compromise the accuracy and sensitivity required for normal operation. In some cases, this will necessitate use of overlapping ranges of instruments to monitor the required range of the variable to be monitored, possibly with different performance requirements in each range.

Standard ANS-4.5,* "Criteria for Accident monitoring Functions in a Light-Water-Cooled Nuclear Power Generating Station," dated ________ 1980, delineates criteria for determining the variables to be monitored by the control room operator, as required for safety, during the course of an accident and during the long-term stable shutdown phase followng an accident. Standard ANS-4.5 was prepared by Working Group 4.5 of Subcommittee ANS-4 with two primary objectives: (1) to address that instrumentation that permits the operator to monitor expected parameter changes in an accident period and (2) to address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events. ANS-4.5 references a revision to IEEE Std 497 as the source for specific instrumentation design criteria. Since the revision to IEEE Std 497 has not yet been completed, its applicability cannot yet be determined. Hence, specific instrumentation design criteria have been included in this regulatory guide.

The ANS standard defines three variable types (definitions modified herein) for the purpose of aiding the designer in his selection of accident-monitoring instrumentation and applicable criteria. The types are: Type A - those variables that provide primary** information needed to permit the control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and which are required for safety systems to accomplish their safety functions for design basis accident events. Type B - those variables that provide information to indicate whether plant safety functions are being accomplished, and Type C - those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release, i.e., fuel cladding, primary coolant pressure boundary, and containment (modified to reflect NRC staff position; see Position C.1.2). The sources of potential breach are limited to the energy sources within the barrier

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*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, LaGrange Park, Illinois 60525.

**Primary information is that which is essential for the direct accomplishment of the specified safety functions and does not include those variables which are associated with contingency actions that may also be identified in written procedures.

itself. In addition to the accident monitoring variables provided in ANS-4.5 standard, variables for monitoring the operation of systems important to safety and radioactive effluent releases are provided by this regulatory guide. Two additional variable types are defined. They are: Type D - those variables that provide information to indicate the operation of individual safety systems and other systems important to safety, and Type E - those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases,

A minimum set of Types B, C, D, and E variables to be measured is listed in this regulatory guide. Type A variables have not been listed because they are plant specific and will depend on the operations that the designer chooses for planned manual action. Types B, C, D, and E are variables for following the course of an accident and are to be used (a) to determine if the plant is responding to the safety measures in operation, (b) to inform the operator of the necessity for unplanned actions to mitigate the consequences of an accident. The five classifications are not mutually exclusive in that a given variable (or instrument) may be applicable to one or more types, as well as for normal power plant operation or for automatically initiated safety actions. A variable included as Type B, C, D, or E does not preclude that variable from being included as Type A also. Where such multiple use occurs, it is essential that instrumentation be capable of meeting the ~~most~~ more stringent requirements.

The time phases (Phases I, and II) delineated in ANS-4.5 are not used in this regulatory guide. These considerations are plant specific. It is important that the required instrumentation survive the accident environment and function as long as the information it provides is needed by the control room operating personnel.

Regulatory Positions C.1.3 and C.1.4 of this guide provide design and qualification criteria for the instrumentation used to measure the various variables listed in Table 1 (for BWR) and Table 2 (for PWR). The criteria are separated into three separate groups or categories which provide a graded approach to requirements depending on the importance to safety of a variable being measured. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 requires less stringent requirements and generally applies to instrumentation designated for indicating system operating status. Category 3 is intended to provide requirements which will assure that high-quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. It is also used where state-of-the-art will not support requirements for higher qualified instrumentation.

In general, the measurement of a single key variable may not be sufficient to indicate the accomplishment of a given safety function. Where multiple variables are needed to indicate the accomplishment of a given safety function, it is essential that they each be considered key variables and measured with high-quality instrumentation. Additionally, it is prudent, in some instances, to include the measurement of additional variables for backup information and for diagnosis. Where these additional measurements are included, the measures applied for design, qualification, and quality assurance of the instrumentation need not be the same as that applied for the instrumentation for key variables. A key variable is that single variable (or minimum number of variables) that most directly indicate the accomplishment of a safety function (in the case of Types B & C) or the operation of a system safety (in the case of Type D) or radioactive materials release (in the case of Type E). It is essential that key variables be qualified to the more stringent design and qualification criteria. The design and qualification criteria category assigned to each variables, indicates whether the variable is considered to be a key variable or for system status indication or for backup or diagnosis, i.e., for Types B and C, the key variables are Category 1; backup variables are generally Category 3. For Types D and E, the key variables are generally Category 2, backup variables are Category 3.

The variables are listed but no mention (beyond redundancy requirements) is made of the number of points of measurement of each variable. It is important that the number of points of measurement be sufficient to adequately indicate the variable value, e.g., containment temperature may require spatial location of several points of measurement.

This guide provides the minimum variables to be monitored by the control room operating personnel during and following an accident. These variables are used by the control room operating personnel to perform their role in the emergency plan in the evaluation, assessment, monitoring, and execution of control room functions when the other emergency response facilities are not effectively manned. Variables are also defined to permit the operator to perform his long-term monitoring and execution responsibilities after the emergency response facilities are manned. The application of the criteria for the instrumentation is limited to that part of the instrumentation system and its vital supporting features or power sources which provide the direct display of the variables. These provisions are not necessarily applicable to that part of the instrumentation systems provided as operator aids for the purpose of enhancement of information presentations for the identification or diagnosis of disturbances.

C. REGULATORY POSITION

1. ACCIDENT MONITORING INSTRUMENTATION

The criteria, and requirements, contained in Standard ANS-4.5,"Criteria for Accident Monitoring Functions in a Light-Water-Cooled Nuclear Power Generating Station," dated _________ 1980, are considered by the NRC staff to be generally acceptable for providing instrumentation to monitor variables for accident conditions subject to the following:

1.1 In Section 3.2.1 of ANS-4.5, the definition of Type A variables should be modified to be as follows: Type A - those variables to be monitored that provide the primary information required to permit the control room operator to take the specified manually controlled actions for which no automatic control is provided and which are required for safety systems to accomplish their safety function for design basis accident events. (Note: Primary information is that which is essential for the direct accomplishment of the specified safety function and does not include those variables which are associated with contingency actions that may also be identified in written procedures.)

1.2 In Section 3.2.3 of ANS-4.5, the definition of "Type C" includes two items, (1) and (2). Item (1) includes those instruments that indicate the extent to which parameters which have the potential for causing a breach in the primary reactor containment have exceeded the design basis values. In conjunction with the parameters that indicate the potential for causing a breach in the primary reactor containment, the parameters that indicate the potential for causing a breach in the fuel cladding (e.g., core exit temperature) and the reactor coolant pressure boundary (e.g., reactor coolant pressure) should also be included. The sources of potential breach are limited to the energy sources within the cladding, coolant boundary or containment. References to Type C instruments, and associated parameters to be measured, in Standard ANS-4.5 (e.g., Sections 4.2, 5.0, 5.1.3, 5.2, 6.0, 6.3) should include this expanded definition.

1.3 Section 6.1 of ANS-4.5 pertains to General Criteria for Types A, B, and C accident monitoring variables. In lieu of Section 6.1, the following design and qualification criteria categories should be used:

1.3.1 Design and Qualification Criteria - Category 1

(1) The instrumentation should be qualified in accordance with the methodology described in Regulatory Guide 1.89 & NUREG 0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. Where the instrumentation channel signal is to be used in a computer-based display, recording and/or diagnostic program, qualification applies from the sensor to and ~~including~~ includes the channel isolation device. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions. The seismic portion of qualification should be in accordance with Regulatory Guide 1.100. Instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake. Instrumentation, whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable, should be qualified using the guidance provided in paragraph 6.3.6 of ANS-4.5.

(2) No single failure within either the accident-monitoring instrumentation, its auxiliary supporting features or its power sources concurrent with the failures that are a condition or result of a specific accident, should prevent the operator from being presented the information necessary for him to determining the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident. Where failure of one accident-monitoring channel results in information ambiguity (that is, the redundant displays disagree) which could lead the operator to defeat or fail to accomplish a required safety function, additional information should be provided to allow the operator to

from each other and from equipment not classified important to safety

deduce the actual conditions in the plant. This may be accomplished by providing additional independent channels of information of the same variable (addition of an identical channel), or by providing an independent channel which monitors a different variable which bears a known relationship to the multiple channels (addition of a diverse channel)~~, or by providing the capability, if sufficient time is available, for the operator to perturb the measured variable and determine which channel has failed by observation of the response on each instrumentation channel.~~ Redundant or diverse channels should be electrically independent and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. At least one channel should be displayed on a direct-indicating or recording device. (NOTE: Within each redundant division of a safety system, redundant monitoring channels are not needed except for steam generator level instrumentation in two-loop plants.)

(3) The instrumentation should be energized from station Standby Power sources as provided in Regulatory Guide 1.32, battery backed where momentary interruption is not tolerable.

(4) The instrumentation channel should be available prior to an accident except as provided in Paragraph 4.11, "Exemption", as defined in IEEE Std 279 or as specified in Technical Specifications.

(5) The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

Regulatory Guide 1.28	"Quality Assurance Program Requirements (Design & Construction)"
Regulatory Guide 1.30	"Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"
Regulatory Guide 1.38	"Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"
Regulatory Guide 1.58	"Qualification of Nuclear Power Plant Insptection, Examination, and Testing Personnel"
Regulatory Guide 1.64	"Quality Assurance Requirements for the Design of Nuclear Power Plants"

Regulatory Guide 1.74	"Quality Assurance Terms and Definitions"
Regulatory Guide 1.88	"Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"
Regulatory Guide 1.123	"Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"
Regulatory Guide 1.144	"Auditing of Quality Assurance Programs for Nuclear Power Plants"
Regulatory Guide 1.146	"Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" ~~(Guide number to be inserted.)~~

Reference to the above regulatory guides (except Regulatory Guides 1.30, and 1.38) are being made pending issuance of a regulatory guide endorsing NQA-1 (Task RS 002-5) which is in progress.

(6) Continuous indication (it may be by recording) display should be provided. Where two or more instruments are needed to cover a particular range, overlapping of instrument spans should be provided.

(7) Recording of instrumentation readout information should be provided. Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on dedicated recorders ~~analog stripchart~~. Otherwise, it may be continuously updated, computer memory stored, and displayed on demand. Intermittent displays, such as data loggers and scanning recorders, may be used if no significant transient response information is likely to be lost by such devices.

1.3.2 Design and Qualification Criteria - Category 2

(1) The instrumentation should be qualified in accordance with Regulatory Guide 1.89 & the methodology described in NUREG 0588. Where the channel signal is to be processed or displayed on demand, qualification applies from the sensor through the isolator/input buffer. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions.

(2) The instrumentation should be energized from a high reliability power source, not necessarily Standby Power, battery backed where momentary interruption is not tolerable.

(3) The out-of-service interval should be based on normal Technical Specification requirements on out-of-service for the system it serves where applicable or where specified by other requirements.

(4) The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

Regulatory Guide 1.28	"Quality Assurance Program Requirements (Design & Construction)"
Regulatory Guide 1.30	"Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"
Regulatory Guide 1.38	"Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"
Regulatory Guide 1.58	"Qualification of Nuclear Power Plant Insptection, Examination, and Testing Personnel"
Regulatory Guide 1.64	"Quality Assurance Requirements for the Design of Nuclear Power Plants"
Regulatory Guide 1.74	"Quality Assurance Terms and Definitions"
Regulatory Guide 1.88	"Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"
Regulatory Guide 1.123	"Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"
Regulatory Guide 1.144	"Auditing of Quality Assurance Programs for Nuclear Power Plants"
<u>Regulatory Guide 1.146</u>	"Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" ~~(Guide number to be inserted.)~~

Reference to the above regulatory guides (except Regulatory Guides 1.30, and 1.38) are being made pending issuance of a regulatory guide endorsing NQA-1 (Task RS 002-5) which is in progress. Since some instrumentation is less important to safety than other instrumentation, it may not be necessary to apply the same quality assurance measures to all instrumentation. The quality assurance requirements, which are implemented, should provide control over activities affecting quality to an extent consistent with the importance to safety of the instrumentation. These requirements should be determined and documented by personnel knowledgeable in the end use of the instrumentation.

(5) The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand by a CRT or other appropriate means.

(6) The method of display may be dial, digital, CRT or stripchart recorder indication. Effluent ~~release monitors should be recorded, including effluent~~ radioactivity monitors, ~~environs exposure rate~~ area radiation monitors, and meteorology monitors should be recorded. Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on dedicated recorders ~~analog stripchart~~. Otherwise, it may be continuously updated, computer memory stored, and displayed on demand.

1.3.3 Design and Qualification Criteria - Category 3

(1) High quality commercial grade instrumentation selected to withstand the specified service environment.

(2) (repeated as above)

1.4 In addition to the criteria of Position C.1.3, the following criteria should apply to Categories 1 and 2:

1.4.1 Any equipment that is used for either Category 1 or Category 2 should be designated as part of accident monitoring or systems operation and effluent monitoring instrumentation. The transmission of signals from such equipment for other use should be through isolation devices that are designated as part of monitoring instrumentation and that meet the provisions of this document.

1.4.2 The instruments designated as Types A, B and C and Categories 1 and 2 should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

1.5 In addition to the above criteria, the following should apply to Categories 1, 2 and 3.

~~1.5.1 Means should be provided for checking, with a high degree of confidence the operational availability of each monitoring channel, including its input sensor, during reactor operation. This may be accomplished in various ways, for example:~~

~~(1) By perturbing the monitored variable;~~

~~(2) By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or~~

~~(3) By cross-checking between channels that bear a known relationship to each other and that have readouts available.~~

1.5.~~2~~1 Servicing, testing, and calibration programs should be specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing will be less than the normal time interval between generating station shutdowns, a capability for testing during power operation should be provided.

1.5.~~3~~2 Whenever means for removing channels from service are included in the design, the design should facilitate administrative control of the access to such removal means.

1.5.~~4~~3 The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

1.5.~~5~~4 The monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator.

1.5.~~6~~5 The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

1.5.~~7~~6 To the extent practicable ~~[practical] possible~~, monitoring instrumentation inputs should be from sensors that directly measure the desired variables. *An indirect measurement should be made only when it can be shown by analysis to provide unambiguous information.*

1.5.~~8~~7 To the extent practicable ~~practical~~, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operator to use, during accident situations, instruments with which he is most familiar. However, where the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used.

1.5.~~9~~8 Periodic *checking,* testing, *calibration, and calibration verification* should be in accordance with the applicable portions of Regulatory Guide 1.118 pertaining to testing of instruments channels. (Note: Response time testing is not usually needed.)

1.6 Sections 6.2.2, 6.2.3, 6.2.4, 6.2.5, 6.2.6, 6.3.2, 6.3.3, 6.3.4, and 6.3.5 of ANS-4.5 pertain to variables and variable ranges for monitoring Types B and C variables. In conjunction with the above sections, Tables 1, and 2 of this regulatory guide (which include those variables mentioned in the above sections) should be used as the minimum set of instruments and their respective ranges for accident-monitoring instrumentation for each nuclear power plant.

2. SYSTEMS OPERATION MONITORING AND EFFLUENT RELEASE MONITORING INSTRUMENTATION

2.1 Definitions

2.1.1 Type D - those variables that provide information to indicate the operation of individual safety systems and other systems important to safety.

2.1.2 Type E - those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

2.2 The plant designer should select variables and information display channels required by his design to enable the control room operating personnel to:

2.2.1 Ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation to help mitigate the consequences of an accident.

2.2.2 Monitor the effluent discharge paths and environs within the site boundary to ascertain if there have been significant releases (planned or unplanned) of radioactive materials and for continually assessing such releases.

2.2.3 Obtain required information through a backup or diagnosis channel where a single channel may be likely to give ambiguous indication.

2.3 The process for selecting system operation and effluent release variables should include the identification of:

2.3.1 For Type D

(1) the plant safety systems and other systems important to safety which should be operating or which could be placed in operation to help mitigate the consequences of an accident;

(2) the variable or minimum list of variables that indicate the operating status of each system identified in (1) above.

2.3.~~3~~2 For Type E

(1) the planned paths for effluent release;

(2) plant areas and inside buildings where access is required to service equipment necessary to mitigate the consequences of an accident;

(3) onsite locations where unplanned releases of radioactive materials should be detected;

(4) the variables that should be monitored in each location identified in (1), (2), and (3) above.

2.4 The determination of performance requirements for system operation monitoring and effluent release monitoring information display channels should include, as a minimum, identification of:

(1) the range of the process variable.
(2) the required accuracy of measurement.
(3) the required response characteristics.
(4) the time interval during which the measurement is needed.
(5) the local environment(s) in which the information display channel components must operate.
(6) any requirement for rate or trend information.
(7) any requirements to group displays of related information.
(8) any required spatial distribution of sensors.

2.5 The design and qualification criteria for system operation monitoring and effluent release monitoring instrumentation should be taken from the criteria provided in Positions C.1.3 and C.1.4 of this guide. Tables 1 and 2 of this regulatory guide should be used as a minimum set of instruments and their respectives ranges for systems operation monitoring (Type D) and effluent release monitoring (Type E) instrumentation for each nuclear power plant.

D. IMPLEMENTATION

All plants going into operation after June ~~1982~~ 1983 should meet the provisions of this guide.

Plants currently operating ~~or scheduled to be licensed to operate before June 1, 1982~~ should meet the requirements of NUREG-0578 and NRR letters dated September 13, and October 30, 1979. The provisions of this guide as specified in Tables 1, and 2 for operating plants are compatible with these documents. ~~which are to be completed by January 1, 1981.~~ *Implementation schedules for these items have been provided in NUREG-0578, NUREG-0660, NUREG-0694, and subsequent NRR letters dated September 13, and October 30, 1979. The implementation schedule shown in NUREG-0737 supersedes previously provided schedules.* The balance of provisions of the guide are to be completed by June 1983.

Plants scheduled to be licensed to operate before June 1, 1983 should meet the requirements of NUREG-0737 according to the schedule provided in NUREG-0737 or prior to the issuance of a license to operate whichever date is later. The balance of provisions of the guide should be completed by June 1983.

The difficulties of procuring and installing additions or modifications to in-place instrumentation have been considered in establishing these schedules.

Exceptions to requirements and schedules will be considered for extraordinary circumstances.

TABLE 1

BWR VARIABLES

Type A - Those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and which are required for safety systems to accomplish their safety function for design basis accident events. Primary information is that which is essential for the direct accomplishment of the specified safety function and does not include those variables which are associated with contingency actions that may also be identified in written procedures.

A variable included as Type A does not preclude it from being included as Type B, C, D, or E, or vice versa.

Variable	Range	Category (see Position C.1.3)	Purpose
Plant specific	plant specific	1	Information required for operator action

TABLE 1

BWR VARIABLES (continued)

TYPE B Variables - Those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, (4) containment integrity (which includes radioactive effluent control). Variables are listed with designated ranges and category for design and qualification requirements. Key variables are indicated by design and qualification Category 1.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE B VARIABLES			
Reactivity Control			
Neutron Flux	10^{-6} % to ~~5%~~ 100% full power (SRM, APRM)	1	Function detection; Accomplishment of mitigation
Control Rod Position	Full in or not full in	3	Verification
RCS Soluble Boron Concentration (Sample)	0 to 1000 ppm	3	Verification
Core Cooling			
Coolant Level in the Reactor	Bottom of core support plate to ~~above the top of discharge plenum~~ lesser of top of vessel or centerline of main steam line	1	Function detection; Accomplishment of mitigation; Long-term surveillance
BWR Core Thermocouples	~~Unresolved~~[5] 200°F to 2300°F	To be determined[5]	~~To monitor core cooling if water level is low, spray is lost, or channels restricted.~~ To provide diverse indication of water level

TABLE 1 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE B - continued			
Maintaining Reactor Coolant System Integrity			
RCS Pressure[1]	15 psia to ~~2000~~ 1500 psig	1 X	Function detection; Accomplishment of mitigation; Verification
Drywell Pressure[1]	0 to design pressure[2] (psig)	1	Function detection; Accomplishment of mitigation; Verification
Drywell Sump Level[1]	Bottom to top	1 X	Function detection; Accomplishment of mitigation; Verification
Maintaining Containment Integrity			
Primary Containment Pressure ~~(Drywell)~~[1]	10 psia to design pressure[2]	1	Function detection; Accomplishment of mitigation; Verification
Primary Containment Isolation Valve Position (excluding check valves)	Closed - not closed	1	Accomplishment of isolation

TABLE 1 (continued)

BWR VARIABLES (continued)

TYPE C Variables - Those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are: (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE C VARIABLES			
Fuel Cladding			
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	½ Tech Spec limit to 100 Times Tech Spec limit, R/hr	1	Detection of breach
~~Accident Sampling and~~ Analysis of Primary Coolant ~~- Gross Activity~~ · Gamma Spectrum	10 µCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume	3[17]	Detail analysis; Accomplishment of mitigation; Verification; Long-term surveillance
BWR Core Thermocouples	~~Unresolved[2]~~ 200°F to 2300°F	To be determined[5]	To monitor core cooling if water level is low, ~~spray is lost, or channels restricted~~
Reactor Coolant Pressure Boundary			
RCS Pressure[1]	15 psia to 1500 psig	1[4]	Detection of potential for or actual breach; Accomplishment of mitigation; Long-term surveillance
Primary Containment Area Radiation[1]	1 R/hr to 10^5 R/hr	3[7, 11]	Detection of breach; Verification
Drywell Drain Sumps[1] Level (Identified and Unidintified Leakage)	Bottom to top	1	Detection of breach; Accomplishment of mitigation; Verification; Long-term surveillance
Suppression Pool Water Level ~~(for operating plants)~~	Bottom of ECCS suction line to 5ft above normal water level	1	Same as immediately above

TABLE 1 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE C - continued			
Reactor Coolant Pressure Boundary (continued)			
Drywell Pressure[1]	0 to design pressure[2] (psig)	1	Detection of breach; Verification
Containment			
RCS Pressure[1]	15 psia to 1500 psig	1[4]	Detection of potential for breach; Accomplishment of mitigation
Primary Containment[1] Pressure (~~Drywell~~)	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel	1	Detection of potential for or actual breach; Accomplishment of mitigation
Containment and Drywell Hydrogen Concentration	0 to 30% (capability of operating from 12 psia to design pressure[2])	1	Detection of potential for breach; Accomplishment of mitigation
Containment and Drywell Oxygen Concentration (for inerted containment plants)	0 to 10% (capability of operating from 12 psia to design pressure[2])	1	Detection of potential for breach; Accomplishment of mitigation
Containment Effluent[1] Radioactivity - Noble Gases (from identified release points including Standby Gas Treatment System Vent)	10^{-6} to 10^{-2} μCi/cc	3[9] [10]	Detection of actual breach; Accomplishment of mitigation; Verification
Environs Radioactivity - Exposure Rate[1]	1mR/hr ~~10~~ to 10 R/hr	3 2[8] ~~11~~	Detection of breach; Accomplishment of mitigation; Verification

TABLE 1 (continued)

BWR VARIABLES (continued)

TYPE D Variables - those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D VARIABLES			
Condensate and Feedwater System			
Main Feedwater Flow	0 to 110% design flow[3]	3	Detection of operation; Analysis of cooling
Condensate Storage Tank Level	Bottom to top	3	Indication of available water for cooling
Primary Containment-Related Systems			
Suppression Chamber Spray Flow	0 to 110% design flow[3]	2	To monitor operation
Drywell Pressure[1]	12 psia to 3 psig 0 to 110% design pressure[2]	2	To monitor operation
Suppression Pool Water Level	Top of vent to top of weir well	2	To monitor operation
Suppression Pool Water Temperature	30°F to 230°F	2	To monitor operation
Drywell Atmosphere Temperature	40°F to 440°F	2	To monitor operation
Drywell Spray Flow	0 to 110% design flow[3]	2	To monitor operation

TABLE 1 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D - continued			
Main Steam System			
~~Main Steamline Flow~~	~~0 to 120% design flow[3]~~	~~1~~	~~To monitor operation~~
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	2 X	To provide indication of pressure boundary maintenance
Primary System Safety Relief Valve Positions, including ADS or Flow Through or Pressure in Valve lines	Closed-not closed or 0 to 50 psig	2 X	Detection of accident; boundary integrity indication

TABLE 1 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D - continued			
Safety Systems			
Isolation Condenser System Shell-side Water Level	Top to bottom	2	To monitor operation
Isolation Condenser System Valve Position	Open or closed	2	To monitor status
RCIC Flow	0 to 110% design flow[3]	2	To monitor operation
HPCI Flow	0 to 110% design flow[3]	2	To monitor operation
Core Spray System Flow	0 to 110% design flow[3]	2	To monitor operation
~~RHR~~ LPCI System Flow ~~(LPCI)~~	0 to 110% design flow[3]	2	To monitor operation
~~RHR Heat Exchanger Outlet Temperature (LPCI)~~	~~32°F to 350°F~~	~~2~~	~~To monitor operation~~
SLCS Flow	0 to 110% design flow[3]	2 ~~3~~	To monitor operation
SLCS Storage Tank Level	Bottom to top	2 ~~3~~	To monitor operation
Residual Heat Removal Systems			
RHR System Flow	0 to 110% design flow[3]	2	To monitor operation
RHR Heat Exchanger Outlet Temperature	32°F to 350°F	2	To monitor operation

TABLE I (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D - continued			
Cooling Water System			
Cooling Water Temperature to ESF System Components	32°F to 200°F	2	To monitor operation
Cooling Water Flow to ESF System Components	0 to 110% design flow[3]	2	To monitor operation
Radwaste Systems			
High Radioactivity Liquid Tank Level	Top to bottom	3	To monitor operation
Ventilation Systems			
Emergency Ventilation Damper Position	Open-closed status	2	To monitor operation
Power Supplies			
Status of Standby Power & Other energy Sources Important to Safety *(hydraulic, pneumatic)*	Voltages, currents, pressures	2[12]	To monitor ~~operation~~ *system status*

TABLE 1 (continued)

BWR VARIABLES (continued)

TYPE E Variables - Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E VARIABLES			
Containment Radiation			
Primary Containment Area Radiation - High Range[1]	1 R/hr to 10^7 R/hr	1[7 11]	Detection of significant releases; Release assessment; Long-term surveillance; Emergency plan actuation
Reactor Bldg or Secondary Containment Area Radiation	10^{-1} R/hr to 10^4 R/hr ~~10^{-5} to 10^4 µCi/cc~~ for Mark 1 and 2 1 R/hr to 10^7 R/hr for Mark 3	2[10] 1[7 11]	Detection of significant releases; Release assessment Long-term surveillance
Area Radiation			
Radiation Exposure Rate (Inside bldgs or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	2[11]	Detection of significant releases; Release assessment; Long-term surveillance
Airborne Radioactive Materials Released from the Plant			
Noble Gases and Vent Flow Rate			
o Drywell Purge, Standby Gas Treatment System Purge (for Mark I, II, III plants) & Secondary Containment Purge (for Mark I plants)	10^{-6} to 10^5 µCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment
o Secondary Containment Purge (for Mark I, II, III plants)	10^{-6} to 10^4 µCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment

TABLE 1 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - continued			
Airborne Radioactive Materials Released from the Plant			
Noble Gases and Vent Flow Rate (continued)			
o Secondary Containment (reactor shield bldg annulus, if in design)	10^{-6} to 10^{4} μCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment
o Auxiliary Building (including any bldg containing primary system gases, e.g., waste gas decay tank)	10^{-6} to 10^{4} μCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment; Long-term surveillance
o Common Plant Vent or Multi-purpose Vent Discharging Any of the Above Releases (If drywell or SGTS purge is included)	10^{-6} to 10^{3} μCi/cc 0 to 110% vent design flow[3] 10^{-6} to 10^{4} μCi/cc	2[10]	Detection of significant releases; Release assessment; Long-term surveillance
o All Other Identified Release Points	10^{-6} to 10^{2} μCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru other monitored plant vents)	2[10]	Detection of significant releases; Release assessment; Long-term surveillance
Particulates and Halogens			
o All Identified Plant Release Points. Sampling, with Onsite Analysis Capability	10^{-3} to 10^{2} μCi/cc 0 to 110% vent design flow[3]	3[13]	Detection of significant releases; Release assessment; Long-term surveillance

TABLE I (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - continued			
Environs Radiation and Radioactivity			
Radiation Exposure Rate[1] (Installed instrumentation)	1mR/hr ~~10^{-6} R/hr~~ to 10 R/hr	3 ~~2~~[8] ~~[11]~~	Detection and estimation of significant releases; Verification; Release assessment; Long-term surveillance
Airborne Radiohalogens and Particulates (portable sampling, with on-site analysis capability)	10^{-9} to 10^{-3} µCi/cc	3[14]	Release assessment; Analysis
Plant and Environs Radiation (Portable Instrumentation)	10^{-3} ~~0.1~~ to 10^{4} R/hr, photons 10^{-3} ~~0.1~~ to 10^{4} rads/hr, beta radiations and low-energy photons	3[15] 3[15]	Release assessment; Analysis
Plant and Environs Radioactivity (Portable Instrumentation)	Multi-channel Gamma-Ray spectrometer	3	Releases assessment; Analysis

POOR ORIGINAL

TABLE I (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - Continued			
METEOROLOGY[16]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1.0 mph). Damping ratio between 0.4 and 0.6, distance constant ≤2 meters.	3	Release assessment
Wind Speed	0 to 30 mps (67 mph) ±0.22 mps (0.5 mph) accuracy for wind speeds less than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1.0 mph).	3	Release assessment
Estimation of Atmosphric Stability	Based on vertical temperature difference from primary system, -5°C to 10°C (-9°F to 18°F) and ±0.15°C accuracy per 50 meter intervals (±0.3°F accuracy per 164 foot intervals) or analogous range for ~~back-up system.~~ alternate stability estimate.	3	Release assessment

POOR ORIGINAL

TABLE 1 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - (continued)			
ACCIDENT SAMPLING CAPABILITY (Analysis Capability Onsite)*			
Primary Coolant & Sump	Grab Sample	3[17,18]	Release assessment; Verification; Analysis
o Gross Activity	10 µCi/ml to 10 Ci/ml		
o Gamma Spectrum	(Isotopic Analysis)		
o Boron Content	0 to 1 000 ppm		
o Chloride Content	0 to 20 ppm		
o Disolved Oxygen	0 to 20 ppm		
o pH	1 to 13		
Containment Air	Grab Sample	3[17]	Release assessment; Verification; Analysis
o Hydrogen Content	0 to 10% 0 to 30% for inerted containments		
o Oxygen Content	0 to 30%		
o Gamma Spectrum	(~~Noble gas~~ Isotopic analysis)		

*The time for taking and analysing samples should be 3 hours or less from the time the decision is made to sample, except chloride which should be within 24 hours.

POOR ORIGINAL

TABLE I (continued)

NOTES

[1]Where a variable is listed for more than one purpose, the instrumentation requirements may be integrated and only one measurement provided.

[2]Design pressure is that value corresponding to ASME code values that are obtained at or below code-allowable material design stress values.

[3]Design flow is the maximum flow anticipated in normal operation.

[4]The maximum value may be revised upward to satisfy ATWS requirements.

[5]~~The number of thermocouples, their range and location to be determined.~~ *Two to four thermocouples per quadrant located in instrument thimbles 1/3 to 1/2 way down from top of core.*

[6]~~Measurement should be made of the gross gamma radiation emanating from circulating primary coolant, with instrument calibration permitting conversion of readout to radioactivity concentrations in terms of either curies/gram or curies/unit volume. System accuracy should be ±½ order of magnitude. The point of measurement should be external to a circulating primary coolant line or loop, and should not be a line or loop subject to isolation, e.g., main steam line. While such an instrument may not be currently available off-the-shelf, the staff considers that the necessary components are available commercially and have been employed and demonstrated under adverse environmental conditions in high-level hot cell operations for many years.~~

[7]Minimum of two monitors at widely separated locations.

[8]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) ~~continuous readout capability. (Approximately 16 to 20 locations - site dependent.)~~

[9]Provisions should be made to monitor all identified pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64. Monitoring of individual effluent streams only is required where such streams are released directly into the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this guide provided such monitoring has a range adequate to measure worst-case releases.

[10]Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fresh equilibrium noble gas fission product mixtures to 10-day old mixtures, with overall system accuracies *within a factor of 2* ~~of ±½ decade~~. Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of any noble gas nuclide(s). It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this guide and that multiple components or systems will be needed. Existing equipment may be utilized to monitor any portion of the stated range within the equipment design rating. ~~Additional extended range instrumentation should overlap the range of existing instrumentation by at least a factor of 2.~~

[11]Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an *energy response* accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within ~~±½ decade~~ *a factor of 2* over the entire range.

NOTES - continued

[12]Status indication of all StandBy Power A-C Buses, D-C buses, inverter output buses and pneumatic supplies.

[13]To provide information regarding release of radioactive halogens and particulates. Continous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentrations of 10^2 μCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^2 μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration.

[14]For estimating release rates of radioactive materials released during an accident. ~~from unidentified release paths (not covered by effluent monitors). Continous collection of representative samples followed by laboratory measurements of the samples. (Approximately 16 to 20 locations - site dependent.)~~

[15]To monitor radiation and airborne radioactivity concentrations in many areas throughout the facility and the site environs where it is impractical to install stationary monitors capable of covering both normal and accident levels.

[16]Meteorological measurements should conform to the provisions of the forthcoming Draft Revision 1 ~~revision~~ of Regulatory Guide 1.23, "~~Onsite~~ Meteorological Programs In Support of Nuclear Power Plants." September 1980.

[17]Sampling or monitoring of radioactive liquids and gases should be performed in a manner which assures procurement of representative samples. For gases, the criteria of ANSI N13.1 should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones and sampling lines should be designed to minimize plateout or deposition. For safe and convenient sampling, the provisions should include:

a. Shielding to maintain radiation doses ALARA,
b. Sample containers with container-sampling port connector compatability,
c. Capability of sampling under primary system pressure and negative pressures,
d. Handling and transport capability, and
e. Pre-arrangement for analysis and interpretation.

[18]An installed capability should be provided for obtaining containment sump, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

TABLE 2

PWR VARIABLES

Type A - Those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and which are required for safety systems to accomplish their safety function for design basis accident events. Primary information is that which is essential for the direct accomplishment of the specified safety function and does not include those variables which are associated with contingency actions that may also be identified in written procedures.

A variable included as Type A does not preclude it from being included as Type B, C, D, or E, or vice versa.

Variable	Range	Category (see Position C.1.3)	Purpose
Plant specific	plant specific	1	Information required for operator action

TABLE 2

PWR VARIABLES (continued)

Type B Variables - Those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, (4) containment integrity (which includes radioactive effluent control). Variables are listed with designated ranges and category for design and qualification requirements. Key variables are indicated by design and qualification Category 1.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE B VARIABLES			
Reactivity Control			
Neutron Flux	10^{-6} % to ~~5%~~ 100% full power	1	Function detection; Accomplishment of mitigation.
Control Rod Position	Full in or not full in	3	Verification
RCS Soluble Boron Concentration	0 to 6000 ppm	3	Verification
RCS Cold Leg Water Temperature[1]	50°F to 400°F	3	Verification
Core Cooling			
RCS Hot Leg Water Temperature	50°F to 750°F	1	Function detection; Accomplishment of mitigation; Verification; Long-term surveillance
RCS Cold Leg Water Temperature[1]	50°F to 750°F	1	Function detection; Accomplishment of mitigation; Verification; Long-term surveillance
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[4]	Function detection; Accomplishment of mitigation; Verification; Long-term surveillance

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE B - continued			
Core Cooling (continued)			
Core Exit Temperature[1]	200° ~~150°~~F to 2300°F (for operating plants - 200° ~~150°~~F to 1650°F)	3[5]	Verification
Coolant Level in Reactor	Bottom of core to top of vessel	~~1~~ (Direct indicating or recording device not needed)	Verification, accomplishment of mitigation
Degrees of Subcooling	200°F subcooling to 35°F superheat	~~1~~ ~~(for operating plants -~~ 2, with confirmatory operator procedures)	Verification and analysis of plant conditions
Maintaining Reactor Coolant System Integrity			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[4]	Function detection; Accomplishment of mitigation
Containment Sump Water[1] Level	Narrow range (sump), Wide range (bottom of containment to 600,000-gallon level equivalent)	3 1	Function detection; Accomplishment of mitigation; Verification
Containment Pressure[1]	0 to design pressure[2] (psig)	1	Function detection; Accomplishment of mitigation; Verification

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE B - continued			
Maintaining Containment Integrity			
Containment Isolation Valve Position (excluding check valves)	Closed-not closed	1	Accomplishment of isolation
Containment Pressure[1]	10 psia to design pressure[2]	1	Function detection Accomplishment of mitigation Verification

TYPE C Variables - Those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are: (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE C VARIABLES			
Fuel Cladding			
Core Exit Temperature[1]	~~150°~~ 200°F to 2300°F (for operating plants - ~~150°~~ 200°F to 1650°F)	1[5]	Detection of potential for breach; Accomplishment of mitigation; Long-term surveillance
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	½ Tech Spec limit to 100 times Tech Spec limit R/hr	1	Detection of breach
~~Accident Sampling and~~ Analysis of Primary Coolant ~~Gross Activity~~ · Gamma Spectrum	10 μCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume	3[18]	Detail analysis; Accomplishment of mitigation; Verification; Long-term surveillance
Reactor Coolant Pressure Boundary			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[4]	Detection of potential for or actual breach; Accomplishment of mitigation; Long-term surveillance
Containment Pressure[1]	10 psia to design pressure[2] psig (5 psia for sub-atmospheric containments)	1	Detection of breach; Accomplishment of mitigation; Verification; Long-term surveillance

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE C - continued			
Reactor Coolant Pressure Boundary (continued)			
Containment Sump Water Level[1]	Narrow range (sump), Wide range (bottom of containment to 600,000-gallon level equivalent)	3 1	Detection of breach; Accomplishment of mitigation; Verification; Long-term surveillance
Containment Area Radiation[1]	1 to 10^4 R/hr	3[7 11]	Detection of breach; Verification
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust[1]	10^{-6} to 10^{-2} μCi/cc	3[10]	Detection of breach; Verification
Containment			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[4]	Detection of potential for breach; Accomplishment of mitigation
Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) 0 to 30% for ice condenser type containment	1	Detection of potential for breach; Accomplishment of mitigation Long-term surveillance
Containment Pressure[1]	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel (5 psia for sub-atmospheric containments)	1	Detection of potential for or actual breach; Accomplishment of mitigation

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
<u>TYPE C - continued</u>			
<u>Containment (continued)</u>			
Containment Effluent Radioactivity - Noble Gases from Identified Release Points[1]	10^{-6} to 10^{-2} μCi/cc	2[9] [10]	Detection of breach; Accomplishment of mitigation; Verification
Environs Radioactivity - Exposure Rate[1]	1 mR/hr ~~10^{-4}~~ to 10 R/hr	3 ~~2~~[8] ~~11~~	Detection of breach; Accomplishment of mitigation; Verification

TABLE 2 (continued)

PWR VARIABLES (continued)

TYPE D Variables - Those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D VARIABLES			
Residual Heat Removal or Decay Heat Removal System			
RHR System Flow	0 to 110% design flow[3]	2	To monitor operation
RHR Heat Exchanger Out Temperature	32°F to 350°F	2	To monitor soperation and for analysis
Safety Injection Systems			
Accumulator Tank Level Level or Pressure	10% to 90% volume 0 to 750 psig	2	To monitor operation
Accumulator Isolation Valve Position	Closed or Open	2	Operation status
Boric Acid Charging Flow	0 to 110% design flow[3]	3	To monitor operation
Flow in HPI System	0 to 110% design flow[3]	2	To monitor operation
Flow in LPI System	0 to 110% design flow[3]	2	To monitor operation
Refueling Water Storage Tank Level	Top to bottom	2	To monitor operation
Primary Coolant System			
Reactor Coolant Pump Status	Motor current	3	To monitor operation

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D - continued			
Primary Coolant System - (continued)			
Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines	Closed-not closed	2	Operation status; to monitor for loss of coolant
Pressurizer Level	Bottom to top	1	To assure proper operation of pressurizer
Pressurizer Heater Status	Electric current	3	To determine operating status
Quench Tank Level	Top to bottom	3	To monitor operation
Quench Tank Temperature	50°F to 750°F	3	To monitor operation
Quench Tank Pressure	0 to design pressure[2]	3	To monitor operation
Secondary System (Steam Generator)			
Steam Generator Level	From tube sheet to separators	~~2 (Category~~ 1 ~~for 2-loop plants)~~	To monitor operation
Steam Generator Pressure	From atmospheric pressure to 20% above the lowest safety valve setting	2	To monitor operation
Safety/Relief Valve Positions or Main Steam Flow	Closed - not closed	2	To monitor operation
Main Feedwater Flow	0 to 110% design flow[3]	3	To monitor operation

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D - continued			
Auxiliary Feedwater or Emergency Feedwater System			
Auxiliary or Emergency Feedwater Flow	0 to 110% design flow[3]	2 (1 for B & W plants)	To monitor operation
Condensate Storage Tank Water Level	Plant specific	1	To ensure water supply for auxiliary feedwater (Can be Category 3 if not primary source of AFW. Then whatever is primary source of AFW should be listed and should be Category 1)
Containment Cooling Systems			
Containment Spray Flow	0 to 110% design flow[3]	2	To monitor operation
Heat Removal By the Containment Fan Heat Removal System	Plant specific	2	To monitor operation
Containment Atmosphere Temperature	40°F to 400°F	3	To indicate accomplishment of cooling
Containment Sump Water Temperature	50°F to 250°F	2	To monitor operation

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE D - continued			
Chemical and Volume Control System			
Makeup Flow - In	0 to 110% design flow[3]	2	To monitor operation
Letdown Flow - Out	0 to 110% design flow[3]	2	To monitor operation
Volume Control Tank Level	Top to bottom	2	To monitor operation
Cooling Water System			
Component Cooling Water Temperature to ESF System Components	32°F to 200°F	2	To monitor operation
Component Cooling Water Flow to ESF System Components	0 to 110% design flow[3]	2	To monitor operation
Radwaste Systems			
High-Level Radioactive Liquid Tank Level	Top to bottom	3	To indicate storage volume.
Radioactive Gas Hold-up Tank Pressure	0 to 150% design pressure[2]	3	To indicate storage capacity
Ventilation Systems			
Emergency Ventilation Damper Position	Open-closed status	2	To indicate damper status
Power Supplies			
Status of Standby Power & Other Energy Sources Important to Safety (hydraulic, pneumatic)	Voltages, currents, pressures	2 [13]	To indicate system status

TABLE 2 (continued)

PWR VARIABLES (continued)

TYPE E Variables - Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E VARIABLES			
Containment Radiation			
Containment Area Radiation - Hi Range[1]	1 R/hr to 10^7 R/hr	1[7,11]	Detection of significant releases; Release assessment; Long-term surveillance; Emergency plan actuation
Area Radiation			
Radiation Exposure Rate (Inside bldgs or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	2[11]	Detection of significant releases; Release assessment; Long-term surveillance
Airborne Radioactive Materials Released from the Plant			
Noble Gases and Vent Flow Rate			
o Containment or Purge Effluent[1]	10^{-6} to 10^5 µCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment
o Secondary Containment (reactor shield bldg annulus, if in design)	10^{-6} to 10^4 µCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment
o Auxiliary Building (including any bldg containing primary system gases, e.g., waste gas decay tank)	10^{-6} to 10^{3} µCi/cc 0 to 110% vent design flow[3] (not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment; Long-term surveillance

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - continued			
Airborne Radioactive Materials Release from the Plant (continued)			
Noble Gases and Vent Flow Rate (continued)			
o Condensor Air Removal System Exhaust[1]	10^{-6} to 10^{5} µCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent)	2[10]	Detection of significant releases; Release assessment
o Common Plant Vent or Multi-purpose Vent Discharging Any of the Above Releases (If containment purge is included)	10^{-6} to 10^{3} µCi/cc 0 to 110% vent design flow[3] 10^{-6} to 10^{4} µCi/cc	2[10]	Detection of significant releases; Release assessment; Long-term surveillance
o Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves	10^{-1} to 10^{3} µCi/cc (Duration of releases in seconds, and mass of steam per unit time)	2[12]	Detection of significant releases; Release assessment
o All Other Identified Release Points	10^{-6} to 10^{2} µCi/cc 0 to 110% vent design flow (Not needed if effluent discharges thru other monitored plant vents)	2[10]	Detection of significant releases; Release assessment; Long-term surveillance
Particulates and Halogens			
o All Identified Plant Release Points (except Steam Generator Safety Relief Valves or Atmospheric Steam Dump Valves and Condensor Air Removal System Exhaust) Sampling, With Onsite Analysis Capability	10^{-3} to 10^{2} µCi/cc 0 to 110% vent design flow[3]	3[14]	Detection of significant releases; Release assessment; Long-term surveillance

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - continued			
Environs Radiation and Radioactivity			
Radiation Exposure Rate[1] (Installed instrumentation)	1 mR/hr ~~10^{-6} R/hr~~ to 10 R/hr	3 ~~2~~[8] ~~11~~	Detection and estimation of significant releases; Verification; Release assessment; Long-term surveillance
Airborne Radiohalogens and Particulates (portable sampling, with on-site analysis capability)	10^{-9} to 10^{-3} μCi/cc	3[15]	Release assessment; Analysis
Plant and Environs Radiation (Portable Instrumentation)	10^{-3} ~~0.1~~ to 10^{4} R/hr, photons 10^{-3} ~~0.1~~ to 10^{4} rads/hr, beta radiations and low-energy photons	3[16] 3[16]	Release assessment; Analysis
Plant and Environs Radioactivity (Portable Instrumentation)	Multi-channel Gamma-Ray spectrometer	3	Releases assessment; Analysis

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - Continued			
METEOROLOGY[17]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1.0 mph). Damping ratio between 0.4 and 0.6, distance constant ≤2 meters.	3	Release assessment
Wind Speed	0 to 30 mps (67 mph) ±0.22 mps (0.5 mph) accuracy for wind speeds less than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1.0 mph).	3	Release assessment
Estimation of Atmospheric Stability	Based on vertical temperature difference from primary system, -5°C to 10°C (-9°F to 18°F) and ±0.15°C accuracy per 50 meter intervals (±0.3°F accuracy per 164 foot intervals) or analogous range for ~~back-up system.~~ alternate stability estimates.	3	Release assessment

TABLE 2 (continued)

Variable	Range	Category (see Position C.1.3)	Purpose
TYPE E - (continued)			
ACCIDENT SAMPLING CAP-* ABILITY (Analysis Capability Onsite)			
Primary Coolant & Sump	Grab Sample	$3^{18\ 19}$	Release assessment; Verification; Analysis
o Gross Activity	10 uCi/ml to 10 Ci/ml		
o Gamma Spectrum	(Isotopic Analysis)		
o Boron Content	0 to 6000 ppm		
o Chloride Content	0 to 20 ppm		
o Disolved Oxygen	0 to 20 ppm		
o pH	1 to 13		
Containment Air	Grab Sample	3^{18}	Release assessment; Verification; Analysis
o Hydrogen Content	0 to 10% 0 to 30% for ice condensors		
o Oxygen Content	0 to 30%		
o Gamma Spectrum	Isotopic (~~Noble gas~~ analysis)		

*The time for taking and analysing samples should be 3 hours or less from the time the decision is made to sample, except chloride which should be within 24 hours.

TABLE 2 (continued)

NOTES

[1]Where a variable is listed for more than one purpose, the instrumentation requirements may be integrated and only one measurement provided.

[2]Design pressure is that value corresponding to ASME code values that are obtained at or below code-allowable material design stress values.

[3]Design flow is the maximum flow anticipated in normal operation.

[4]The maximum value may be revised upward to satisfy ATWS requirements.

[5]A minimum of 4 measurements per quadrant is required for operation. Sufficient number should be installed to account for attrition. (Replacement instrumentation should meet the 2300°F range provision.)

[6]~~Measurement should be made of the gross gamma radiation emanating from circulating primary coolant, with instrument calibration permitting conversion of readout to radioactivity concentrations in terms of either curies/gram or curies/unit volume. System accuracy should be ±½ order of magnitude. The point of measurement should be external to a circulating primary coolant line or loop, such as a hot leg, and should not be a line or loop subject to isolation, e.g., letdown line. While such an instrument may not be currently available off the shelf, the staff considers that the necessary components are available commercially and have been employed and demonstrated under adverse environmental conditions in high-level hot cell operations for many years.~~

[7]Minimum of two monitors at widely separated locations.

[8]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - ~~continuous readout capability. (Approximately 16 to 20 locations - site dependent.)~~

[9]Provisions should be made to monitor all identified pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64. Monitoring of individual effluent streams only is required where such streams are released directly into the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this guide provided such monitoring has a range adequate to measure worst-case releases.

[10]Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fresh equilibrium noble gas fission product mixtures to 10-day old mixtures, with overall system accuracies within a factor of 2 ~~of ±½ decade.~~ Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of any noble gas nuclide(s). It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this guide and that multiple components or systems will be needed. Existing equipment may be utilized to monitor any portion of the stated range within the equipment design rating. ~~Additional extended range instrumentation should overlap the range of existing instrumentation by a least a factor of 2.~~

[11]Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an energy response accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within ~~±½ decade~~ a factor of 2 over the entire range.

TABLE 2 (continued)

NOTES - continued

[12]Effluent monitors for PWR steam safety valve discharges and atmospheric steam dump valve discharges should be capable of approximately linear response to gamma radiation photons with energies from approximately 0.5 MeV to 3 MeV. Overall system accuracy should be within a factor of 2 ~~±½ order of magnitude~~. Calibration sources should fall within the range of approximately 0.5 MeV to 1.5 MeV (examples: Cs-137, Mn-54, Na-22, and Co-60). Effluent concentrations should be expressed in terms of any gamma-emitting noble gas nuclide within the specified energy range. Calculational methods should be provided for estimating concurrent releases of low-energy noble gases which cannot be detected or measured by the methods or techniques employed for monitoring.

[13]Status indication of all Standby Power A-C buses, D-C buses, inverter output buses and pneumatic supplies.

[14]To provide information regarding release of radioactive halogens and particulates. Continous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentrations of 10^2 μCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^2 μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration.

[15]For estimating release rates of radioactive materials released during an accident. ~~from unidentified release paths (not covered by effluent monitors). Continous collection of representative samples followed by laboratory measurements of the samples. (Approximately 16 to 20 locations - site dependent.)~~

[16]To monitor radiation and airborne radioactivity concentrations in many areas throughout the facility and the site environs where it is impractical to install stationary monitors capable of covering both normal and accident levels.

[17]Meteorological measurements should conform to the provisions of the forthcoming revision of Regulatory Guide 1.23, "Onsite Meteorological Programs".

[18]Sampling or monitoring of radioactive liquids and gases should be performed in a manner which assures procurement of representative samples. For gases, the criteria of ANSI N13.1 should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones and sampling lines should be designed to minimize plateout or deposition. For safe and convenient sampling, the provisions should include:

a. Shielding to maintain radiation doses ALARA,
b. Sample containers with container-sampling port connector compatability,
c. Capability of sampling under primary system pressure and negative pressures,
d. Handling and transport capability, and
e. Pre-arrangement for analysis and interpretation.

[19]An installed capability should be provided for obtaining containment sump, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

The applicant (licensee) of a nuclear power plant is required by the Commission's regulations to provide instrumentation to (1) monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety and (2) monitor the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents. This revision to Regulatory Guide 1.97 proposes to improve the guidance for plant and environs monitoring during and following an accident, including extended ranges for some instruments to account for consideration of degraded core cooling.

1.2 Need

Regulatory Guide 1.97 was issued as an effective guide in August 1977. At the time the guide was issued, it was recognized that more specific guidance than that contained in the guide would be required. However, the difficulty in developing the guide to the point where it could be initially issued was evidence that experience in using the guide as it then existed was essential before further development of the guide would be meaningful.

Therefore, in August 1977, the staff initiated Task Action Plan A-34, "Instruments for Monitoring Radiation and Process Variables During an Accident." The purpose of the task action plan was to develop guidance for applicants, licensees, and staff reviewers concerning implementation of Regulatory Guide 1.97. Such effort would provide a basis for revising the guide.

When the staff was ready to issue the results of the Task Action Plan A-34 effort, the accident at TMI-2 occurred. Subsequently, the TMI-2 Lessons Learned Task Force has issued its "Status Report and Short-Term Recommendations," NUREG-0578.

This report, along with the draft Task Action Plan A-34 report; Draft 1 of Regulatory Guide 1.97, dated April 12, 1974; and Standard ANS-4.5, provides ample basis for revising Regulatory Guide 1.97.

1.3 Value/Impact of the Proposed Action

1.3.1 NRC Operations

Since a list of selected variables to be provided with instrumentation to be monitored by the plant operator during and following an accident has not been explicitly agreed to in the past, the proposed action should result in more effective effort by the staff in reviewing applications for construction permits and operating licenses. The proposed action will establish an NRC position by taking advantage of previous staff effort (1) in completion of a generic activity (A-34), (2) in evaluating the lessons learned from the TMI-2 event (NUREG-0578), and (3) in conjunction with effort in developing a national standard (ANS-4.5). For future plants, the staff review will be simplified with guidance contained in the endorsed standard developed by a voluntary standards group and the regulatory guide, which includes a list of variables for accident monitoring. Efforts by the staff to implement Revision 1 to Regulatory Guide 1.97 has been fraught with frustration and met with delays because the guide was adjudged by licensees to be vague and ambiguous. Revision 2 eliminates the problems encountered with Revision 1 because it provides a minimum set of variables to be measured and hence gives more guidance in the selection of accident monitoring instrumentation. Consequently, there will be no significant impact on the staff. There will, however, be effort required to review each operating plant and plants under construction to ~~assure compliance~~ *assess conformance* with Regulatory Guide 1.97.

1.3.2 Other Government Agencies

Not applicable, unless the government agency is an applicant.

1.3.3 Industry

The proposed action establishes a more clearly defined NRC position with regard to instrumentation to assess plant and environs conditions during and following an accident and, therefore, reduces uncertainty as to what the staff considers acceptable in the area of accident monitoring. Most of the impact on industry will be in the area of providing instrumentation to indicate the potential breach and the actual breach of the barriers to radioactivity release, i.e., fuel cladding, reactor coolant pressure boundary, and containment. These instruments have extended ranges and there are others with qualification requirements not previously imposed. There will be additional impact due to a heretofore unspecified variables to be monitored (i.e., water level in reactor for PWRs and radiation level in the primary coolant water for PWRs and BWRs) that have been identified during the evaluation of TMI-2 experience and will require development.

Attempts were made during the comment period to determine the cost impact on industry for future plants and for backfitting existing plants. Estimates ranged from $4,000,000 to over $20,000,000. The higher estimates undoubtedly charged all accident monitoring instrumentation to Revision 2 to Regulatory Guide 1.97 which should not be the case. The requirement for accident monitoring has always been a part of the regulations. Consequently the impact of Revision 2 to Regulatory Guide 1.97 should only be the delta added by Revision 2. A conservative estimate of the increase in requirements are the additions of Type C measurements and the upgrading of some of the Type B measurements to higher qualification of the instrumentation. There are 17 unique Type B & C variables to be measured for PWRs, less for BWRs. A conservative average cost for each measurement is $130,000 making a total cost impact of $2,210,000. If this figure were doubled to account for overhead costs and about a 15% contingency added, the cost impact would be about $5,000,000. This cost estimate is the same for operating plants as for plants under construction and future plants. While it is recognized that for operating plants the costs associated with backfitting are generally higher than the costs associated with new plants, there are some concessions made in some of the requirements due to existing licensing committments which brings the cost estimate to about the same value.

1.3.4 Public

The proposed action will improve public safety by ensuring that the plant operator will have timely information to take any necessary action to protect the public.

No impact on the public can be foreseen.

1.4 Decision on Proposed Action

As previously stated, more definitive guidance on instrumentation to assess plant and environs conditions during and following an accident should be given.

2. TECHNICAL APPROACH

This section is not applicable to this value/impact statement since the proposed action is a revision of an existing regulatory guide, and there are no alternatives to providing the plant operator with the required information.

3. PROCEDURAL APPROACH

Previously discussed.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Authority for this guide would be derived from the safety requirements of the Atomic Energy Act through the Commission's regulations, in particular, Criterion 13, Criterion 19, and Criterion 64 of Appendix A to 10 CFR Part 50, which require, in part, that instrumentation be provided to monitor variables, systems, and plant environs to ensure adequate safety.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined in paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No conflicts or overlaps with requirements promulgated by other agencies are foreseen. This guide does include the variables to be monitored on site by the plant operator in order to provide necessary information for emergency planning. However, emergency planning and its relationship to other agencies is provided by other means. Implementation of the proposed action is discussed in Section D of the proposed revision.

6. SUMMARY AND CONCLUSIONS

The revision to Regulatory Guide 1.97, "Instrumentation For Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," should be issued and implemented according to existing schedules.

AMERICAN NUCLEAR SOCIETY
STANDARDS COMMITTEE

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October 1, 1980
LS-80-161

TO: ANS/NUPPSCO Members
IEEE/NPEC Members

SUBJECT: Industry Review of NRC Regulatory Guide 1.97

Enclosed are detailed comments on this Guide prepared last week by the ANS 4.5 Writing Group. The comments address (1) design and qualification category criteria, and (2) specific variables, ranges, and applicable criteria for both BWR and PWR plants.

Review and discussion of this material is scheduled at the October 15th NUPPSCO and the October 29th NPEC meetings. Formal submittal of approved concensus comments to NRC is planned.

While these comments have been prepared using a Working Paper F draft of the Guide dated September 9th, its technical content is virtually identical with the previous June 2nd version submitted to ACRS and a subsequent Working Paper E version dated August 20th. You should be able to secure a copy of one of these versions within your company organization prior to the NUPPSCO and NPEC meetings. The only significant changes in these versions are:

(1) NRC has reduced the scope to that of the control room operator;

(2) Table 1 criteria has been converted to text format, and

(3) Design and Qualification Categories 1, 2, and 3 have been established to specify applicable requirements to specific variables.

In a separate effort, an AIF Task Force chaired by Bill Coley has developed variable recommendations to "facilitate management of the accident situation" in the control room, the TSC, the EOF, and on the SPDS described in NUREG-0696. These recommendations are enclosed for consideration at the NUPPSCO and NPEC meetings.

I look forward to reviewing this material with you at the meetings, and greatly appreciate your assistance in providing a thorough review of this Regulatory Guide.

Sincerely,

Loren Stanley

Loren Stanley
Chairman, ANS 4.5

LS/bjk
Enclosures
cc: W. Coley, P. Higgins (AIF)
ANS 4.5 Writing Group

POOR ORIGINAL

EMERGENCY RESPONSE FACILITY ACCIDENT MONITORING

I. ACCIDENT MANAGEMENT FUNCTIONS

The emergency response facilities/display systems consist of the control room, safety parameter display system (SPDS), the technical support center (TSC), the emergency operations facility (EOF), and the nuclear data link (NDL). These facilities/systems will operate as integrated system to enhance management and control of plant emergency response capabilities. These facilities are designed to provide a graduated response to emergencies as determined by the severity of the accident and the time following accident initiation.

The approach used to select variables for each emergency facility requires careful definition of the functions to be carried out in managing an accident and the particular instrumentation requirements associated with each function. In this context, variables will be interpreted as real-time "data" needs in order to distinguish these from off-line background "information". (Actual decision-making in an accident will depend more upon "information", which results from an off-line analysis of the data, than from the availability of real-time raw data, in and of itself.)

In this section the accident management functions and variable selection criteria are described. In the following section (Section II) the role of emergency facilities on an accident are outlined in accordance with their assigned accident functions. This systematic approach has established the basis for a comprehensive parameter selection exercise which addresses the needs for all of the emergency facilities. The parameter lists which were generated using this approach are contained in attachments 1 through 6.

In Section III the results of a validation study is described in which event tree analysis and emergency operator quidelines were used to confirm the adequacy of the forementioned parameter lists.

Detection

Detection applies to early phases of an event, providing early warning or confirmation to operators that an operating transient or accident sequence has begun. Variables selected for this function are those which are the most direct and reliable indicators of the approach to nuclear plant operating or safety limits. These variables are, however, less directly associated with the possible cause or causes of an event, which may have to do with the failure of particular piece(s) of equipment or maloperation of equipment. Detection is the first function that must be fulfilled in the sequence of actions taken to manage an accident. Detection variables are of high safety significant and must be: reliable, environmentally qualified, redundant, independent and sufficiently robust that their failure will not prevent the initiation of required safety functions. These variables must be supported by diagnostic capability. Variables selected in this document for detection are those for which any of the following conditions apply:

a. Parameter is a leading indicator for the initiating event in a dominant accident sequence.
b. Breach or failure of a radioactive barrier is directly indicated.
c. Parameter is used in reactor protection (e.g., input variable for reactor trip); the signal may in general be equivalent to that used for reactor trip, but not be the same sensor or have the same range.

Mitigation Feedback

Mitigation feedback is the means by which operations and support personnel have the capability to assess the correctness of event diagnosis and the effectiveness of manual and automatic actions that are taken. This function presupposes event detection and initial response. However, no presumption is made regarding the adequacy of the response or whether the event falls within the confines of accident pre-analysis. The mitigation feedback function does not rely on variables which describe the operability of individual safety systems, but is strongly linked to the overall plant response to the operation of these safety systems. Variables used for mitigation feedback must be reliable, environmentally qualified, rudundant, independent, and supported by diagnostic capability. Variables in this document are selected for mitigation feedback if they directly responsive to plant copnditions relative any of the following critical safety functions:

a. reactivity
b. coolant inventory
c. heat removal
d. containment integrity
e. radiological conditions.

Within the context of the safety parameter display system (SPDS) mitigation feedback variables are the key parameters which are most directly reponsive to the accomplishment of safety functions. Mitigation feedback variables for the control room (at large) also include secondary and back-up variables which bear upon or add depth to an assessment of these functions.

Validation/Diagnosis

Validation/diagnosis is a defense-in-depth function which permits independent confirmation of a detection or mitigation feedback variable, and provides alternate variables should this become necessary in an accident situation (e.g., sensor failure for a primary variable). Variables selected for validation/diagnosis may be useful as confirmatory indicators either singley or in combination with other validation/diagnosis variables. In general, these variables are not expected to validate primary indicators in a numerical sense, but provide reliable information as to their response and correctness in trend. Validation/diagnosis variables are less safety significant than variables used for detection and mitigation feedback. Redundancy and independence are not required; sensor environmental qualification should be considered only for in-containment or other potentially hostile environment locations. Variables in this document are selected for validation/diagnosis if they are required (separately or in combination with other diagnostic/validation variable) to confirm the behavior of a detection or mitigation feedback parameter.

Event Evaluation/Analysis

Event evaluation/analysis is a function whereby the state of the plant, inclusive of all safety and important to safety subsystems is characterized. This function is dependent upon accident time frame and the particular emergency facility involved. In the control room, analysis is principally related to determination of root causes(s) and confirmation that safety, and important-to-safety, subsystems are operating satisfactorily. Actuation of the technical support center provides additional analysis capability in order to confirm or assist in identification of root causes(s), but also provide the means for: assessing the nuclear and thermal-hydraulic state of the plant, estimating damage to fuel and radioactive barriers; determining actual or projected consequences of releases in terms of public exposure. Analysis variables are less safety significant than detection and mitigation feedback variables. Redundancy and independence are not required. Seismic qualification should be considered for all analysis variables, environmental qualification (other than seismic) is required for instruments in locations which may be inaccessible during an accident. Variables in this document are selected for analysis for which any of the following conditions apply:

a. Parameter is an indicator of safety or important-to-safety systems performance.
b. Parameter is needed to perform primary heat or mass balances.
c. Parameter may be used to evaluate core reactivity.
d. Variable provides information as to the physical condition of the core.
e. Parameter is needed to evaluate the state and effectiveness of radioactive barriers.

Radiological Releases

The radiological release function is directed to the determination of the actual or projected consequences of an accident in terms of public radiological exposure or risk. Distinction must be made between in-plant releases which effect operations and radiological events or conditions which produce or may lead to off-site exposures. It is the latter concern that covered by this function. The scope covers planned as well as unplanned radioactive releases, and includes the assessment of the potential for release (e.g., challenge of containment building integrity) as well as evaluation of the release itself. Radiological release instrumentation must be redundant in the sense that successive barriers to radioactive release must be properly monitored. These variables should primarily serve the purpose of detection rather than quantitative evaluation. Seismic qualification should be required for all such permanently installed instrumentation. Portable and off-line instrumentation with sampling capacility should be provided for identification of specific radionuclides and quantitative assessment of release(s). Variables in this document are selected for detection for which any of the following conditions apply:

a. Parameter detects primary (and secondary) coolant activity.
b. Parameter indicates potential for breaching containment building.
c. Parameter detects the presence of radiation in plant spaces.
d. Parameter detects radiation at plant effluent points.
e. Parameter monitors environs radiation at site boundary and off-site locations.

Long Term Surveillance

Long term surveillance applies to the (possible) extended post-accident period in which the plant condition is maintened in a stable, but off-normal mode pending recovery.

Identification of variables required for long term surveillance is dependent upon the specific accident or event. It is anticipated that informational needs will be influenced by revised operating procedures and temporary equipment/instrumentation which may be installed to cope with particular plant circumstances. As a minimum, the state of critical safety functions (i.e., primary system coolant inventory, heat removal, core criticality, containment integrity, and radiological conditions) must be continually monitored over the long term. A basic set of plant variables must be identified and subjected to the most stringent qualification requirements to assure that no interruption in monitoring capability occurs despite the presence of a prolonged, adverse environment in some plant areas. Variables in this document are selected for long-term surveillance for which all the following conditions apply:

a. Accessibility to the snnsor following an accident would probably be limited and an alternative measurement cannot be installed (in the post-accident environment) to support long term surveillance.
b. Parameter is considered to be of importance to long term surveillance of most postulated events or accidents (i.e., related to status of critical safety functions).

II. ROLES AND FUNCTIONS OF EMERGENCY RESPONSE FACILITIES

The staged response concept to facility activation and utilization requires that certain primary roles and functions will shift from the control room to the TSC and EOF in order to enhance emergency response management. Table 1 illustrates this integrated and staged approach to assigning principal and supporting roles to the emergency response facilities. During the initial stages of an accident, all principal roles are assigned to the control room until TSC activation. Following TSC activation, the principal role for diagnosis, analysis, and radiological release assessment is shifted from the control room to TSC. Following EOF activation, the principal role for radiological release assessment is shifted to the EOF.

A. Control Room

1. Role

- o Detect and determine root cause(s) for the accident.
- o Conduct real-time analysis of plant conditions and

trends.

- Mitigate accidents by accomplishing critical safety functions and pre-planned manual actions.
- Detect or assess the potential for radioactivity release to the environs.
- Initiate site emergency plan.
- Terminate the accident.
- Continuous and long-term surveillance of critical safety functions and required safety systems.

2. Functions

Detection

- Reliable and direct measurements indicative of an accident condition or event.
- Reliable and direct leading indicators for analyzed accident sequences

Mitigation Feedback

- Variables to follow the course of accident events indicative of the accomplishment or maintenance of critical safety functions.

- Variables necessary for undertaking pre-planned manual actions.
- Variables descriptive of the effect of accident mitigation actions whether automatically or manually initiated.
- Variables for monitoring the barriers to radioactivity release.

Validation Diagnosis

- Variables that provide functional diversity in the detection of an accident event and in the mitigation feedback to assess the accomplishment or maintaining of critical safety functions.

Event Evaluation Analysis

- Variables to monitor the operation of plant safety systems under accident conditions.
- Variables to assist the control room operator in determining the cause of the accident.
- Historical trend data where required.

Radiological Releases

- Radioactive effluent monitoring variables from identified plant release points.
- Variables for indicating the potential of significant (gross) radioactive releases to the environs (i.e., containment radioactivity)
- Variables indicative of an actual breach of primary reactor containment.

Long-Term Surveillance

- Variables to assess the accomplishing or maintaining of critical safety functions during the long-term recovery period.

- Variables that indicate the operation of required safety systems during the long-term recovery period.

B. Safety Parameter Display System

1. Role

   - o Provide continuous reliable indication of key plant variables representative of critical safety functions.
   - o Concentrate key plant variables in a compact display format.
   - o Serve as an operator's aid.
   - o Provide information feedback to operators to determine whether the overall plant condition is trending in a safe direction.

2. Function

   Detection

   - o Monitor key variables indicative of an accident condition or event.
   - o Provide reliable and direct measurements under accident conditions.

   Mitigation Feedback

   - o Key variables used to follow the course of accident events indicative of accomplishing or maintaining critical safety functions.

   Validation Diagnosis

   - o No role

   Event Evaluation Analysis

   - o No role

   Radiological Releases

   - o No role

   Long-Term Surveillance

   - o Key variables to assess the accomplishing or maintaining of critical safety functions during the long-term recovery period.

C. Technical Support Center

1. Role

- o Manage the plant emergency response.
- o Provide advice and guidance to control room operating staff.
- o Monitor the accomplishment of critical safety functions.
- o Responsible for diagnostics to assure valid data.
- o Conduct detailed off-line analysis of plant conditions and trends to support and augment control room analyses.
- o Provide validated information of the EOF and other off-site facilities.
- o Serve as primary communication link to control room for obtaining additional plant data.
- o Archive data for post-accident assessment and plant recovery.

2. Functions

Detection

- o No role

Mitigation Feedback

- o Key variables used to follow the course of accident events indicative of the accomplishing and maintaining of critical safety functions.
- o Information feedback on the effect of automatic or manual actions to mitigate the accident.

Validation Diagnosis

- o Redundant or diverse information on key variables to validate data used in the mitigation feedback function .
- o Provision for reasonable access to plant information and facilities to validate other variables, as required.

Event Evaluation Analysis

- o Variables which can be used to assess core conditions.
- o Variables to evaluate conditions of radioactive barriers.
- o Information to assist or confirm control room identification of event cause(s).
- o Information to monitor performance of important-to-safety systems.
- o Information to predict the magnitude of controlled radiological releases, as required.

Radiological Releases

- o Radioactive effluent monitoring variables from identified plant release points.
- o Variables for indicating the potential for significant (gross) releases to the environs.
- o Variables indicative of an actual breach of primary reactor containment.

Long-Term Surveillance

- o No role

D. Emergency Operations Facility

1. Role

- o Manage overall site emergency response.
- o Evaluate the magnitude and extent of radiological releases.
- o Recommend appropriate off-site protective measures.
- o Coordinate emergency response with local, state, and federal agencies.

2. Functions

Detection

- o No role

Mitigation Feedback

- o No role

Validation Diagnosis

- o No role

Event Evaluation Analysis

- o Radiological and meteorological information to permit evaluation of the magnitude and extent of radiological releases.

Radiological Release Assessment

- o Information sufficient to assess the environmental consequences of radiological releases.

Long-Term Surveillance

o No role

III VALIDATION OF PARAMETER LISTS

To be provided later.

TABLE 1

EMERGENCY MANAGEMENT FUNCTIONS

ERF System Status	TSC and EOF Not Activated		EOF Not Activated			All Facilities/Systems Activated				
FUNCTION \ FACILITY	Control Room	SPDS	Control Room	SPDS	TSC	Control Room	SPDS	TSC	EOF	NDL
DETECTION	P	S	P	S	N	P	S	N	N	N
MITIGATION FEEDBACK	P	S	P	S	S	P	S	S	N	S
DATA VALIDATION	P	N	S	N	P	S	N	P	N	N
EVENT EVALUATION ANALYSIS	P	N	S	N	P	S	N	P	N	N
RADIOLOGICAL RELEASE ASSESSMENT	P	N	S	N	P	S	N	S	P	N
LONG TERM SURVEILLANCE	P	S	P	S	N	P	S	N	N	S

Key: P = Principal Role
S = Supporting Role
N = No Role

ATTACHMENT 1

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -
PWR

Variable	FUNCTIONS Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
Core Exit temperature	x	x	x	x		x
Hot leg temperature	x	x	x	x		x
Cold leg temperature	x	x	x	x		x
Reactor Coolant Pressure	x	x	x	x		x
Neutron Flux-Source	x	x	x	x		x
Neutron Flux-Intermediate	x	x	x	x		
Vessel liquid level			x	x		
Reactor Coolant Activity	x	x		x	x	x(sample)
Subcooled Margin			x	x		
Control Rods Positions			x	x		
Pressurizer level	x	x	x	x		x
Boron Concentration			x	x		x(sample)
Containment pressure	x	x		x	x	x
Containment activity	x	x		x	x	x(sample)
Containment Hydrogen	x	x		x	x	x(sample)
Containment Sump level	x	x	x	x		x
Containment isolation valve Pos.*		x		x	x	
Pressurizer S/R Valve Pos/Flow				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -
PWR

Variable	FUNCTIONS					
	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
Steam Generator liquid level	x	x		x		
Steam Generator Pressure	x	x		x		
Aux Building Vent Gross Gamma				x	x	
Gross Gamma @ primary release	x	x			x	
Ventilation System Emergency Vent Damper Pos.**		x		x	x	
Environs Radiation Monitor			x		x	
LPI ESF Demand				x		
BWST Liquid Level				x		
Reactor Bldg. Cooling ESF Demand				x		
Control Rod Breaker Status	x	x				
Reactor Protective System Trip Demand	x					
Containment Temperature			x	x		x
Emergency Power Availability				x		
Letdown Flow Rate			x	x		
Makeup Flow Rate			x	x		
PZR Heater Status				x		
Quench Tank Level				x		
Quench Tank Temperature				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

- CONTROL ROOM -

PWR

	FUNCTIONS					
Variable	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
Steam Generator liquid level	x	x		x		
Steam Generator Pressure	x	x		x		
Aux Building Vent Gross Gamma				x	x	
Gross Gamma @ primary release	x	x			x	
Ventilation System Emergency Vent Damper Pos.**		x		x	x	
Environs Radiation Monitor			x		x	
LPI ESF Demand				x		
BWST Liquid Level				x		
Reactor Bldg. Cooling ESF Demand				x		
Control Rod Breaker Status	x	x				
Reactor Protective System Trip Demand	x					
Containment Temperature			x	x		x
Emergency Power Availability				x		
Letdown Flow Rate			x	x		
Makeup Flow Rate			x	x		
PZR Heater Status				x		
Quench Tank Level				x		
Quench Tank Temperature				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -
PWR

Variable	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
			FUNCTIONS			
Quench Tank Pressure				x		
RC Pump Status or Flow				x		
Core Flood Tank Level			x	x		
Core Flood Tank Pressure				x		
HPI emand				x		
HPI Discharge Flow			x	x		
Volume Control Tank Level			x	x		
LPI Discharge Flow			x	x		
RHR Inlet and Outlet Temp.			x	x		
Main Feedwater Pump Flow!			x	x		
Emergency Feedwater Pump Demand				x		
Emergency Feedwater Pump Flow			x	x		
Secondary Side Relief Valve Pos.			x	x		
Condenser Off-Gas Gross Gamma				x	x	
Condensate Storage Tanks Level*				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -
PWR

Variable	FUNCTIONS Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
Reactor Building Spray Pump Demand				x		
Spray Pump Discharge Flow				x		
RB Cooling Fan Demand				x		
Heat Removed by RB Coolers			x	x		
Area Radiation Monitors				x		
Wind Speed					x	
Wind Direction					x	

* If source or emergency feedwater
** One channel of indication required for each valve, since valves are redundant

ATTACHEMNT 2

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

SAFETY PARAMETER DISPLAY PANEL

PWR

Variable	Detection	Mitigation Feedback	FUNCTIONS			Long-Term Surveillance
Neutron Flux (<1% Power)	X	X				X
RCS Cold Leg Temp	X	X				X
RCS Hot Leg Temp or Core Exit Temp	X	X				X
RCS Pressure	X	X				X
Pressurizer Water Level	X	X				X
Steam Generator Water Level	X	X				
Steam Generator Pressure	X	X				
Auxiliary Feedwater Flow	X	X				
Main Feedwater Flow	X	X				
Containment Pressure	X	X				X
Containment High-Range Area Radiation	X	X				X (SAMPLE)
Containment Sump Water Level	X	X				X
Secondary Side Radiation (Air Ejector Off-Gas)	X	X				
Stack Radioactivity Noble	X	X				

... REQUIREMENTS FOR ACCIDENT MANAGEMENT

TECHNICAL SUPPORT CENTER
PWR

FUNCTIONS

Variable		Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	
Core Exit Temperature		X	X	X		
Hot Leg Temperature		X	X	X		
Cold Leg Temperature		X	X	X		
Reactor Coolant Pressure		X	X	X		
Neutron Flux-Source		X	X	X		
Vessel Liqu ' Level			X	X		
Reactor Coolant Activity		X		X	X	
Pressurizer Level		X	X	X		
Boron Concentration			X	X		
Containment Pressure		X		X	X	
Containment Activity		X		X	X	
Containment Hydrogen		X		X	X	
Containment Sump Level		X	X	X		
Steam Generator Liquid Lev.		X		X		
Steam Generator Pressure		X		X		
Aux. Building Vent Gross Gamma				X	X	
Gross Gamma @ Primary Release Points		X			X	
Environs Radiation Monitor			X	X	X	

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
TECHNICAL SUPPORT CENTER
PWR

		FUNCTIONS				
Variable		Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	
BWST Liquid Level				X		
Containment Temperature			X	X		
Letdown Flow Rate			X	X		
Makeup Flow Rate			X	X		
Quench Tank Level				X		
Quench Tank Pressure				X		
RC Pump Status or Flow				X		
HPI Discharge Flow			X	X		
Volume Control Tank Level			X	X		
LPI Discharge Flow			X	X		
RHR Inlet and Outlet Temp.			X	X		
Main Feedwater Pump Flow			X	X		
Emergency Feedwater Pump Flow			X	X		
Gross Gamma in Area of Steam Relief and Vent Valves			X	X	X	
Condensor Off-Gas Activity				X	X	
Condensate Storage Tanks Level*				X		
Reactor Building Spray Flow				X		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
TECHNICAL SUPPORT CENTER
PWR

Variable		FUNCTIONS Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	
Heat Removed by RB Coolers			X	X		
Area Radiation Monitors				X		
Wind Speed					X	
Wind Direction					X	

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -

BWR

Variable	FUNCTIONS					
	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
Reactor Coolant Pressure	x	x	x	x		x
Neutron Flux -APRM	x	x	x	x		
Neutron Flux - Source	x	x	x	x		x
Vessel Liquid Level	x	x	x	x		x
Reactor Coolant Activity	x	x		x	x	x (sample)
Reactor Protection System Demand	x	x				
Dry Well Pressure	x	x		x	x	x
Containment Activity	x	x		x	x	x (sample)
Containment Hydrogen Concentration If not Inerted	x	x		x	x	x (sample)
Containment Oxygen Concentration If enerted	x	x		x	x	x (sample)
Dry well Sump Level	x	x	x	x		
Primary Containment Isolation Valves Position **		x		x	x	
Gross Gamma @ Primary Release Pts	x	x		x	x	
Ventilation System Exhaust Damper Position **		x		x	x	
Primary System Leak Detection Outside Containment	x	x				
Primary System Isolation Valve Position **		x		x	x	
Secondary Containment Isolation Valves Position **		x		x	x	
MSIV Leakage Control Inlet Pressure		x				
Plant Ventilation Activity Monitors				x	x	
Wetwell Pressure				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -

BWR

Variable	FUNCTIONS					
	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
Control Rod Positions			x	x		
Dry Well Temperature			x	x		
Safety/Relief Valve Position				x		
Environ Radiation Monitor			x		x	
Emergency Power Availability				x		
Condensate Storage Tank Level*				x		
Area Radiation Monitors				x		
Wind Speed					x	
Wind Direction					x	
Suppression Pool Level			x	x		
Suppression Pool Temperature			x	x		
Secondary Containment Pressure			x	x	x	
Low Pressure Coolant Injection Demand				x		
ADS Demand				x		
LPCI Flow			x	x		
Drywell Spray Flow				x		
Wetwell Spray Flow				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -

BWR

Variable	FUNCTIONS					
	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
RHR Primary Side Flow				x		
RHR Primary Side temp in & out				x		
Wetwell Air Space Temp.			x	x		
Wetwell Air Space Pressure			x	x		
RHR Flow Suppression Pool			x	x		
RHR Flow from Reactor			x	x		
Steam Flow to RHR Hx			x	x		
Core Spray Demand				x		
Core Spray Flow			x	x		
RCIC Demand				x		
RCIC Flow			x	x		
RCIC Steam Flow				x		
HPCI Demand				x		
HPCI Flow			x	x		
HPCI Steam Flow Rate				x		
CRP Cooling Flow			x	x		
Feedwater System Flow			x	x		
Condensor Off Gas Activity				x	x	

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT
- CONTROL ROOM -
BWR

Variable	Detection	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	Long-Term Surveillance
	FUNCTIONS					
Standby Liquid Control Flow			x	x		
Standby Gas Treatment Flow rate				x	x	
Standby Gas Treatment Activity In and Out				x	x	
Building Area Radiation Monitors				x	x	
RWCU System Let Down Flow			x	x		

* If source of emergency feedwater
** One channel of indication required for each valve, since valves are redundant

POOR ORIGINAL

ATTACHMENT 5

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

SAFETY PARAMETER DISPLAY PANEL

BWR

Variable	FUNCTIONS					
	Detections	Mitigation Feedback				Long-Term Surveillance
Vessel Liquid Level	X	X				X
Reactor Coolant Activity	X	X				X (sample)
Condenses Off-Gas Activity	X	X				
Neutron Flux-Source	X	X				X
Reactor Coolant Pressure	X	X				X
Drywell Pressure	X	X				X
Drywell Sump Collection Rate	X	X				
Primary System Isolation Valve Position	X	X				
Safety Relief Valve Position	X	X				
Containment Pressure *	X	X				
Containment Isolation Valve Position	X	X				
Containment Hydrogen Core**	X	X				X (sample)
Supression Pool Level	X	X				X
Supression Pool Level	X	X				
Drywell Temperature	X	X				X
Gross gamma @ primary release points	X	X				

* For Mark III, II and I containment designs these variables are containment/drywell pressure, drywell/wetwell pressure and drywell/torus pressure respectively.

** For inerted containments, containment oxygen concentration.

ATTACHMENT 6

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

TECHNICAL SUPPORT CENTER

BWR

Variable	FUNCTIONS	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	
RHR Primary Side Flow			x	x		
RHR Primary Side temp in & out			x	x		
RHR Flow Suppression Pool			x	x		
RHR Flow from Reactor			x	x		
Steam Flow to RHR Hx			x	x		
Core Spray Flow			x	x		
RCIC Flow			x	x		
RCIC Steam Flow			x	x		
HPCI Flow			x	x		
HPCI Steam Flow Rate			x	x		
CRD Cooling Flow			x	x		
Feedwater System Flow			x	x		
Condensor Off Gas Activity				x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

TECHNICAL SUPPORT CENTER

BWR

		FUNCTIONS				
Variable		Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	
Dry Well Temperature			x	x		
Environ Radiation Monitor			x	x	x	
Condensate Storage Tank Level				x		
Area Radiation Monitors				x	x	
Wind Speed				x	x	
Wind Direction				x	x	
Suppression Pool Level		x	x	x		
Suppression Pool Temperature			x	x		
LPCI Flow			x	x		
Dry Well Spray Flow			x	x		
Wet Well Spray Flow			x	x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

TECHNICAL SUPPORT CENTER

BWR

		FUNCTIONS				
Variable	—	Mitigation Feedback	Diagnosis	Analysis	Radiological Releases	
Dry Well Temperature			x	x		
Environ Radiation Monitor			x	x	x	
Condensate Storage Tank Level				x		
Area Radiation Monitors				x	x	
Wind Speed				x	x	
Wind Direction				x	x	
Suppression Pool Level		x	x	x		
Suppression Pool Temperature			x	x		
LPCI Flow			x	x		
Dry Well Spray Flow			x	x		
Wet Well Spray Flow			x	x		

DATA REQUIREMENTS FOR ACCIDENT MANAGEMENT

TECHNICAL SUPPORT CENTER

BWR

Variable		Mitigation Feedback	Diagnosis	FUNCTIONS Analysis	Radiological Releases	
Standby Liquid Control Flow			x	x		
Standby Gas Treatment Flow Rate				x	x	
Standby Gas Treatment Activity In and Out				x	x	
Building Area Radiation Monitors				x	x	
RWCU System Let Down Flow			x	x		

Comments on Design and Qualification Criteria
Regulatory Guide 1.97, Working Paper F, Sections 1.3.1 through 1.5

1. Section 1.3.1 (1) OK

   Seismic qualification of display or recording device portions of a Category 1 type channel should be applicable to one of the two redundant or diverse channels. This will permit flexibility in using CRT or other devices for the other channel to improve the human factors display to the control room operator.

2. Section 1.3.1 (2) OK

   Physical separation of redundant or diverse Category 1 channels should only be applicable from the sensor to the location of an appropriate isolator in the channel. Once isolation has been accomplished, the requirements of R.G. 1.75 should not be applicable.

3. Section 1.3.1 (6) and (7) OK

   Continuous indication of both redundant or diverse Category 1 channels should not be required at all times. Continuously updated computer memory storage should be permitted for one channel, with its display presented on demand.

4. Section 1.3.2 (6) OK

   For Category 2 variables, analog stripchart recording of trend or transient information should be required only where such recordings are of direct and immediate use to the control room operator. In all other cases, CRT display of stored trend or transient information should be encouraged.

5. Section 1.4.1 Partially

   Category 1 equipment should comply with these classification and isolation requirements; however, since Category 2 equipment is generally used to indicate system operating status, application of these same requirements should not be required. Special control room identification of Category 2 displays should not be required.

6. Section 1.4.10 No

   Category 1 and Category 2 sensors requiring periodic testing, and especially sensor response time testing, should be determined on a variable-by-variable basis consistent with the need for such testing. A blanket requirement for all such variables should be avoided.

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 19 TABLE 2 PAGE 32 FUNCTION: Reactivity Control VARIABLE: Neutron Flux RANGE: 1 C/S to 1% power QUAL. CAT.: 1	PURPOSE: Detection and accident mitigation feedback information (Type B variable) FUNCTION: VARIABLE: RANGE: 10^{-8} to 5×10^{-2} rated power QUAL. CAT.:	JUSTIFICATION: 1 C/S is below a minimum count rate; 10^{-8} power indicates reactor is subcritical; 5% power indicates reactor is critical; beyond 5% power is unnecessary. OK	
TABLE 1 PAGE 19 TABLE 2 PAGE 32 FUNCTION: Reactivity Control VARIABLE: Control Rod Position RANGE: Full in or not full in QUAL. CAT.: 3 (for 1 hour minimum) 3 (for 2 hour minimum)	PURPOSE: Validation information for neutron flux (Type B) FUNCTION: VARIABLE: RANGE: QUAL. CAT.: Delete 1 or 2 hour note	JUSTIFICATION: Operator will verify that all rods are in within first minute; validation data for reactivity control is not needed beyond that point. OK	
TABLE 1 PAGE 19 TABLE PAGE FUNCTION: Core Cooling VARIABLE: Coolant level in the reactor RANGE: Bot. plate to top plenum QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, and long term surveillance information (Type B) FUNCTION: VARIABLE: RANGE: Delete range statement QUAL. CAT.:	JUSTIFICATION: Range is an unresolved generic item at this time. GE-NRC discussions of range are currently in process. No	
TABLE 1 PAGE 19 TABLE PAGE FUNCTION: Core Cooling VARIABLE: Main Steamline Flow RANGE: 0 to 120% design flow QUAL. CAT.: 1	PURPOSE; System status Information (Type D) FUNCTION: Change to System analysis VARIABLE: RANGE: QUAL. CAT.: 3	JUSTIFICATION: MSIV flow is not used for core cooling function; but can be used to confirm accomplishment of containment isolation safety action. OK	
TABLE 1 PAGE 20 TABLE PAGE FUNCTION: Maintaining RCS Integrity VARIABLE: RCS Pressure RANGE: 15 psia to 1500 psig QUAL. CAT.: 1^3	PURPOSE: Detection, accident mitigation feedback, and core cooling validation information (Type B) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 20 TABLE __ PAGE __ FUNCTION: Maintaining RCS Integrity VARIABLE: MSIV Leak. Cntrl Sys. Press. RANGE: 0 to 15" of water, 0 to 5 psid QUAL. CAT.: 1	PURPOSE: System Status Information (Type D) FUNCTION: Change to System Analysis VARIABLE: RANGE: QUAL. CAT.: 3	JUSTIFICATION: MSIV-LCS pressure can confirm system operation but is a very indirect measurement relative to RCPB integrity function OK Cat 1 remains	
TABLE 1 PAGE 20 TABLE __ PAGE __ FUNCTION: Maintaining RCS Integrity VARIABLE: Primary Sys SRV Positions RANGE: Closed-not closed or 0 to 50 psig QUAL. CAT.: 1	PURPOSE: System Status Information (Type D) FUNCTION: Change to System Analysis VARIABLE: RANGE: QUAL. CAT.: 3	JUSTIFICATION: Monitoring each RCPB effluent path for valve position is excessive in determining RCPB integrity; SRV position is useful in analysis of loss of integrity. OK Cat 1 still	
TABLE 1 PAGE 20 TABLE __ PAGE __ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback and RCS integrity validation information (Type B) FUNCTION: RCS Integrity VARIABLE: Drywell Pressure RANGE: 0 psig to drywell design pressure QUAL. CAT.: 1	JUSTIFICATION: This variable should be listed under RCS integrity function with a range up to the drywell design pressure. OK	
TABLE 1 PAGE 20 TABLE __ PAGE __ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback and RCS integrity validation information (Type B) FUNCTION: RCS Integrity VARIABLE: Drywell Sump Level RANGE: (Unresolved Item) QUAL. CAT.: 1	JUSTIFICATION: This variable should be listed under RCS integrity function. The range of level is still unresolved, and requires GE-NRC resolution. OK	
TABLE 1 PAGE 20 TABLE 2 PAGE 34 FUNCTION: Maintaining Cont. Integrity VARIABLE: Prim. Cont. Press. (Drywell) RANGE: 10 psia to 3 x D.P., 4 x D.P. QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, and containment integrity validation information (Type B) FUNCTION: VARIABLE: RANGE: 10 psia to cont. des. pressure QUAL. CAT.:	JUSTIFICATION: For this function, extended range is not necessary beyond the design pressure for containment. Extended range barrier monitoring is a Type C function. OK	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 20 TABLE ___ PAGE ___ FUNCTION: Maintaining Cont. Integrity VARIABLE: Cont. & Drywell Hydrogen Conc. RANGE: 0 to 30% QUAL. CAT.: 1	PURPOSE: Detection and accident mitigation feedback for potential breach of containment (Type C) FUNCTION: Potential Barrier Breach VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: This variable is more properly a Type C rather than a Type B variable. It is an indicator of a potential for breach of containment.	
TABLE 1 PAGE 20 TABLE ___ PAGE ___ FUNCTION: Maintaining Cont. Integrity VARIABLE: Cont. & Drywell Oxygen Conc. RANGE: 0 to 20% QUAL. CAT.: 1	PURPOSE: Detection and accident mitigation feedback for potential breach of containment (Type C) FUNCTION: Potential Barrier Breach VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: This variable is more properly a Type C rather than a Type B variable. It is an indicator of a potential for breach of containment.	
TABLE 1 PAGE 20 TABLE 2 PAGE 34 FUNCTION: Maintaining Cont. Integrity VARIABLE: Prim. Cont Isol. Vlv Pos. (excluding check valves) RANGE: Closed-not closed QUAL. CAT.: 1	PURPOSE: Accident Mitigation feedback information (Type B) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: Clarification that one information channel per valve is required would be beneficial.	
TABLE 1 PAGE 20 TABLE ___ PAGE ___ FUNCTION: Maintaining Cont. Integrity VARIABLE: Supp. Chamber Air Temp. RANGE: 30°F to 230°F QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: Delete Function VARIABLE: Delete Variable RANGE: QUAL. CAT.:	JUSTIFICATION: No containment integrity purpose can be determined or justified for this variable. Deletion is recommended.	
TABLE 1 PAGE 20 TABLE ___ PAGE ___ FUNCTION: Maintaining Cont. Integrity VARIABLE: Drywell Temperature RANGE: 40°F to 440°F QUAL. CAT.: 1	PURPOSE: System Status Information (Type D) FUNCTION: Change to System Analysis VARIABLE: RANGE: QUAL. CAT.: 3	JUSTIFICATION: No containment integrity purpose can be determined or justified for this variable. Its use as a system status indicator could be beneficial.	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: Fuel Cladding VARIABLE: Core Exit Temp RANGE: 150°F to 2300°F (to 1650°F op plt)* QUAL. CAT.: 1^4	PURPOSE: Not determined FUNCTION: Fuel Clad Potential Breach VARIABLE: Delete variable RANGE: QUAL. CAT.: .	JUSTIFICATION: This is a generic issue between GE and NRC. Deletion is recommended until the usefulness of this variable for this purpose is determined. Also consistency with ANS 4.5. Regis 2 T/C	
TABLE 1 PAGE 21 TABLE 2 PAGE 35 FUNCTION: Fuel Cladding VARIABLE: Rad Conc or Rad Lvl in Coolant RANGE: Normal to 10 Ci/gm QUAL. CAT.: 3^5	PURPOSE: Detection of actual breach of the fuel clad barrier (Type C) FUNCTION: Fuel Clad Breach VARIABLE: RANGE: Revise Upper 10 Ci/gm limit QUAL. CAT.: 1	JUSTIFICATION: Value of upper range (10 Ci/gm) should be reduced by using ANS 4.5 range (0.5 to 100 times Tech Spec limit). Type C variable for detecting fuel clad breach should be Category 1. OK	
TABLE 1 PAGE 21 TABLE 2 PAGE 35 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Accident mitigation feedback, validation, analysis, and long-term surveillance information (Type C) FUNCTION: Fuel Clad Breach VARIABLE: Accid. Sample of RCS coolant RANGE: 10 μCi/gm to equiv. to TID-14844 source term in coolant volume QUAL. CAT.: 3	JUSTIFICATION: RCS coolant sample provides validation information for fuel clad breach monitoring variable. OK	
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: RC Pressure Boundary VARIABLE: RCS Pressure RANGE: QUAL. CAT.:	PURPOSE: Not determined FUNCTION: RCPB Potential Breach VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: This variable should be verified as the key variable for determining the potential breach of the RCPB. In the interim, recommend deletion to be consistent with ANS 4.5. No	
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: RC Pressure Boundary VARIABLE: Cont High Range Area Rad. RANGE: 1 to 10^7 R/hr QUAL. CAT.: $1^{6\ 10}$ (Note)	PURPOSE: Detection and validation of RCPB breach function (Type C) FUNCTION: RCPB Breach VARIABLE: RANGE: 1 to 10^5 R/hr QUAL. CAT.: 3	JUSTIFICATION: Range limit should be lowered to remove excessive conservatism and to be consistent with this function. Validation is the primary purpose; hence, cat. 3 is consistent with this objective. Function is actual RCPB breach. OK	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: RC Pressure Boundary VARIABLE: Drywell Drain Sumps Level RANGE: Bottom to Top QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, validation, and long-term surveillance information (Type C) FUNCTION: RCPB Breach VARIABLE: RANGE: (Unresolved item) QUAL. CAT.:	JUSTIFICATION: The range for sump level is an unresolved item. GE-NRC resolution is required. Function is actual breach of RCPB. No	
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: RC Pressure Boundary VARIABLE: Cont. Water Level RANGE: 0 to 5ft above normal level QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, validation, and long term surveillance information (Type C) FUNCTION: RCPB Breach VARIABLE: Suppression Pool Water Level RANGE: QUAL. CAT.:	JUSTIFICATION: Variable name change only. Function is actual breach of RCPB. OK	
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.: 1	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: RCPB Breach VARIABLE: RCS Pressure RANGE: 0 to 1500 psig QUAL. CAT.: 1	JUSTIFICATION: Function is actual breach of RCPB for this variable. OK	
TABLE 1 PAGE 21 TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, validation, and long term surveillance information (Type C) FUNCTION: RCPB Breach VARIABLE: Drywell Pressure RANGE: 10 psia to cont. design press. QUAL. CAT.: 1	JUSTIFICATION: This variable should be listed under RCPB breach Type C with range up to the containment design pressure. OK	
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: Containment VARIABLE: RCS Pressure RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Cont. Potential Breach VARIABLE: RANGE: 0 to 1500 psig QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: Containment VARIABLE: Prim. Cont. Press. (Drywell) RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: RANGE: 10 psia to 3 x D.P. or 4 x D.P. QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: Containment VARIABLE: Cont. & Drywell Hydrogen Conc RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: RANGE: 0 to 30% (with page 20 note) QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: Containment VARIABLE: Cont & Drywell Oxygen Conc. RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: RANGE: 0 to 20% (with page 20 note) QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and analysis information (Type C) FUNCTION: Containment Potential Breach VARIABLE: Suppression Pool Water Temp. RANGE: 30°F to 230°F QUAL. CAT.: 2[3]	JUSTIFICATION: Function is potential breach of containment for this variable. Variable provides information regarding safety system performance also. OK	
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and analysis information (Type C) FUNCTION: Containment Potential Breach VARIABLE: Suppression Pool Water Level RANGE: 0 to 5 feet above normal level QUAL. CAT.: 2[3]	JUSTIFICATION: Function is potential breach of containment for this variable. Variable provides information regarding safety system performance also. OK	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) * FUNCTION: Containment Breach VARIABLE: Prim. Cont. Press. (Drywell) RANGE: 10 psia to 3 x D.P. or 4 x D.P. QUAL. CAT.: 1	JUSTIFICATION: Function is actual breach of containment for this variable as well as potential breach of containment. OK	
TABLE 1 PAGE 22 TABLE 2 PAGE 36 FUNCTION: Containment VARIABLE: Effluent Rad Noble Gases RANGE: 10^{-6} to 10^{5} μCi/cc QUAL. CAT.: $2^{8\ 9}$	PURPOSE: Detection, accident mitigation feedback, and validation information (Type C) FUNCTION: Containment Breach VARIABLE: (See Justification) RANGE: 10^{-6} to 10^{-2} μCi/cc QUAL. CAT.: 3	JUSTIFICATION: Name should be Eff. Rad. - Noble Gases from containment through identified release points including SGTS vent. Lower range is recommended for this function as extended range is provided by Type E variable. OK	
TABLE 1 PAGE 22 TABLE 2 PAGE 36 FUNCTION: Containment VARIABLE: Environs Rad Exposure Rate RANGE: 10^{-6} to 10 R/hr QUAL. CAT.: $2^{7\ 10}$	PURPOSE: Detection, accident mitigation feedback, and validation information (Type C) FUNCTION: Containment Breach VARIABLE: RANGE: 10^{-4} to 10^{1} R/hr QUAL. CAT.:	JUSTIFICATION: Higher lower bound is recommended for this function as extended range is provided by Type E variable. no	
TABLE 1 PAGE 22 TABLE ___ PAGE ___ FUNCTION: Containment VARIABLE: SGTS Vent RANGE: 10^{-6} to 10^{5} μCi/cc QUAL. CAT.: 2^{9}	PURPOSE: Detection, accident mitigation feedback, and validation information (Type C) FUNCTION: Containment Breach VARIABLE: Delete Variable RANGE: QUAL. CAT.:	JUSTIFICATION: This identified release point is included in the Eff. Rad. - Noble Gases from containment monitoring. OK in part	
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Power Conversion Systems VARIABLE: Main Feedwater Flow RANGE: 0 to 110% design flow QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Power Conversion Systems VARIABLE: Condensate Storage Tank Level RANGE: Bottom to Top QUAL. CAT.: 3	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE 1 PAGE 23 TABLE 2 PAGE 37 FUNCTION: Containment Systems VARIABLE: Supp. Pool/Cont Spray Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Containment Systems VARIABLE: Drywell Pressure RANGE: 12 psia to 3 psig; 0 to 110% D.P QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Containment Systems VARIABLE: Supp Chmbr Wtr Lvl RANGE: Top of vent to top of weir wall QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Containment Systems VARIABLE: Supp. Chamber Water Temp. RANGE: 30°F to 230°F QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Auxiliary Systems VARIABLE: CRD Sys. Return Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: No longer required FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Recent design change eliminates CRD return flow. No	
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Auxiliary Systems VARIABLE: Steam Flow to RCIC RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: RCIC System status is determined by output pump flow; the need for inlet steam flow is tenu[illegible] no	
TABLE 1 PAGE 23 TABLE ___ PAGE ___ FUNCTION: Auxiliary Systems VARIABLE: HPCI Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE ___ PAGE ___ FUNCTION: Auxiliary Systems VARIABLE: RCIC Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE ___ PAGE ___ FUNCTION: Auxiliary Systems VARIABLE: Core Spray Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 24 TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: RHR System Flow (LPCI) RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: RHR HX Outlet Temp. (LPCI) RANGE: 32°F to 350°F QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE PAGE FUNCTION: Auxiliary Systems VARIABLE: Service Cooling Water Temp. RANGE: 32°F to 200°F QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE PAGE FUNCTION: Auxiliary System VARIABLE: Service Cooling Water Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE 2 PAGE 38 FUNCTION: Auxiliary System VARIABLE: Flow in UHS Loop RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Need to monitor flow to ultimate heat sink is questioned. Variable deleted	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 24 TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Temp. in UHS Loop RANGE: 30°F to 150°F QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Need to monitor temperature in ultimate heat sink is questioned.	
TABLE 1 PAGE 24 TABLE 2 PAGE 39 FUNCTION: Auxiliary Systems VARIABLE: UHS Level RANGE: Plant Specific QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Need to monitor level of water in ultimate heat sink is questioned.	
TABLE 1 PAGE 24 TABLE PAGE FUNCTION: Auxiliary Systems VARIABLE: SLCS Storage Tank Level RANGE: Bottom to Top QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 24 TABLE 2 PAGE 39 FUNCTION: Auxiliary Systems VARIABLE: Sump level in equip. spaces RANGE: To Level of equip. failure QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Validity of sump measurement to provide meaningful data on probable equipment failure needs to be determined. This is very plant specific.	
TABLE 1 PAGE 24 TABLE PAGE FUNCTION: Auxiliary Systems VARIABLE: SLCS Flow RANGE: 0 to 110% design flow QUAL. CAT.: 3	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 25 TABLE 2 PAGE 39 FUNCTION: Radwaste Systems VARIABLE: High Rad Liquid Tank Level RANGE: Top to Bottom QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 25 TABLE 2 PAGE 39 FUNCTION: Ventilation Systems VARIABLE: Emerg. Vent Damper Position RANGE: Open-closed status QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 25 TABLE 2 PAGE 39 FUNCTION: Ventilation Systems VARIABLE: Temp in Vicinity of Equip. RANGE: 30°F to 180°F QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: Validity of space temperature measurement to accurately predict probable equipment failure needs to be determined. This is very plant specific. Deleted	
TABLE 1 PAGE 25 TABLE 2 PAGE 39 FUNCTION: Power Supplies VARIABLE: Status of Class 1E Sources RANGE: Voltages, currents, pressures QUAL. CAT.: 2[11]	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 25 TABLE 2 PAGE 39 FUNCTION: Power Supplies VARIABLE: Status of non 1E Power RANGE: Voltages, currents, pressures QUAL. CAT.: 3[12]	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Need to monitor non-1E power sources is questioned. deleted	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Effluent Rad Noble Gases RANGE: QUAL. CAT.:	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Entry appears duplicative with noble gas monitoring at specific release points listed below OK	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Environs Rad -Exposure Rate RANGE: QUAL. CAT.:	PURPOSE: Radioactive effluent assessment for others (Type E) FUNCTION: VARIABLE: RANGE: 10^{-4} to 10^{1} R/hr QUAL. CAT.: 2	JUSTIFICATION: ✓	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Gaseous Effl. Flow Rate RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Radioactive effluent assessment by control room operator and others (Type E) FUNCTION: VARIABLE: Merge with Rad Mon variables RANGE: QUAL. CAT.:	JUSTIFICATION: Vent flow rate at point corresponding to radiation measurement is needed; a separate line entry for gaseous effluent flow rate is ambiguous. OK	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and radioactive effluent assessment by control room operator (Type E) FUNCTION: Airborne Rad Mat. Released VARIABLE: Prim.Cont. High Rge Area Rad RANGE: 1 to 10^{7} R/hr QUAL. CAT.: 1	JUSTIFICATION: ✓	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: RX Bldg or Secondary Cont. RANGE: 10^{-6} to 10^{4} μCi/cc QUAL. CAT.: 2^{9}	PURPOSE: Detection and radioactive effluent assessment by control room operator (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: If effluent is monitored downstream, this variable is not required separately. Corresponding flow range is 0 to 110% of design flow. OK	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (8-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Other Release Points RANGE: 10^{-8} to 10^{2} μCi/cc QUAL. CAT.: 2[9]	PURPOSE: Detection and radioactive effluent assessment by control room operator (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: Corresponding flow range is 0 to 110% of design flow. OK	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and radioactive effluent assessment by control room operator (Type E) FUNCTION: Airborne Rad. Mat. Released VARIABLE: Containment Effluent RANGE: 10^{-6} to 10^{5} μCi/cc QUAL. CAT.: 2	JUSTIFICATION: Upper range value is plant specific; lower maximum values may be acceptable. OK	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Eff. Rad Radiohalogens & Part. RANGE: 10^{-3} to 10^{2} μCi/cc QUAL. CAT.: 2[13]	PURPOSE: Radioactive effluent assessment by others (Type E) FUNCTION: VARIABLE: Add (sampling) to name RANGE: QUAL. CAT.:	JUSTIFICATION: L	
TABLE 1 PAGE 26 TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Env. Rad Radiohalogens RANGE: 10^{-9} to 10^{-3} μCi/cc QUAL. CAT.: 2[14]	PURPOSE: Radioactive effluent analysis by others (Type E) FUNCTION: VARIABLE: Add (sampling) to name RANGE: QUAL. CAT.: 3	JUSTIFICATION: Primary function is for analysis; Category 3 seems more appropriate. OK	
TABLE 1 PAGE 27 TABLE 2 PAGE 41 FUNCTION: Rad Rates Inside Buildings VARIABLE: Radiation Exposure Rates RANGE: 10^{-1} R/h to 10^{4} R/hr QUAL. CAT.: 2[10]	PURPOSE: Radioactive effluent analysis by others (Type E) FUNCTION: VARIABLE: RANGE: 10^{-1} mR/hr to 10^{4} mR/hr QUAL. CAT.: 3	JUSTIFICATION: 10^{4} R/hr upper limit is not necessary for analyzing radiological risk to plant personnel. No	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 27 TABLE 2 PAGE 41 FUNCTION: Rad Rates Inside Buildings VARIABLE: Pnt & Env. Rad (portable) RANGE: 0.1 to 10^4 R/hr, 0.1 to 10^4 rads/hr QUAL. CAT.: 3[15]	PURPOSE: Radioactive effluent analysis by others (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 27 TABLE 2 PAGE 41 FUNCTION: Rad Rates Inside Buildings VARIABLE: Plt and Env Rad RANGE: Multi-channel spectrometer QUAL. CAT.: 3	PURPOSE: Radioactive effluent analysis by others (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 27 TABLE 2 PAGE 41 FUNCTION: Meteorology VARIABLE: Wind Direction RANGE: 0 to 360° QUAL. CAT.: 3	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 27 TABLE 2 PAGE 41 FUNCTION: Meteorology VARIABLE: Wind Speed RANGE: 0 to 30 mps QUAL. CAT.: 3	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE 1 PAGE 27 TABLE 2 PAGE 41 FUNCTION: Meteorology VARIABLE: Est. of Atmos Stability RANGE: Based on vert. temp diff. QUAL. CAT.: 3	PURPOSE: Radioactive effluent assessment by others (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	

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NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE 1 PAGE 28 TABLE 2 PAGE 42 FUNCTION: Sampling Capability (Onsite) VARIABLE: Primary Coolant Sampling RANGE: Grab Sample QUAL. CAT.: 3[17 18]	PURPOSE: Radioactive effluent assessment by others (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: OK	
TABLE 1 PAGE 28 TABLE 2 PAGE 42 FUNCTION: Sampling Capability (Onsite) VARIABLE: Primary Sump Sampling RANGE: Grab Sample QUAL. CAT.: 3[17]	PURPOSE: Radioactive effluent assessment by others (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: OK	
TABLE 1 PAGE 28 TABLE 2 PAGE 42 FUNCTION: Sampling Capability (Onsite) VARIABLE: Cont. Air Sampling RANGE: 0 to 30% QUAL. CAT.:	PURPOSE: Radioactive effluent assessment by others (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: OK	
TABLE ___ PAGE ___ TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: Reactivity Control VARIABLE: Soluble Boron Content RANGE: 0 to 6000 ppm QUAL. CAT.: 3	PURPOSE: Validation of reactivity control neutron flux variable (Type B) FUNCTION: VARIABLE: Sol. Boron Conc. in RCS RANGE: Delete note QUAL. CAT.:	JUSTIFICATION: Continuous indication is not meaningful for a periodic sampling and analysis procedure. OK	

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TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: Reactivity Control VARIABLE: Boric Acid Charging Flow RANGE: 0 to 110% design flow QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Charging flow seems unnecessary given neutron flux key variable and RCS boron concentration measurement moved to D	
TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Validation of reactivity control neutron flux variable (Type B) FUNCTION: Reactivity Control VARIABLE: RCS Cold leg temp. RANGE: 50°F to 400°F QUAL. CAT.: 3	JUSTIFICATION: Temperature range chosen as appropriate for reactivity control validation purpose. OK ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: Core Cooling VARIABLE: RCS Hot Leg Temp. RANGE: 50°F to 750°F QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, validation, and long term surveillance information (Type B) FUNCTION: VARIABLE: Add (one per loop) to name RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: Core Cooling VARIABLE: Reactor Coolant Level RANGE: QUAL. CAT.: 1	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Very significant R&D effort required; Reg. Guide should not precede state-of-the-art implementation. No	
TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: Core Cooling VARIABLE: RCS Cold Leg Temp RANGE: 50°F to 750°F QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, validation, and long term surveillance information (Type B) FUNCTION: VARIABLE: Add (one per loop) to name RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	

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TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, validation, and long-term surveillance information (Type B) FUNCTION: Core Cooling VARIABLE: RCS Pressure RANGE: 0 to 3000 psig QUAL. CAT.: 1	JUSTIFICATION: Variable should be added for core cooling function. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 32 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, and long term surveillance information (Type B) FUNCTION: Core Cooling VARIABLE: Core exit temperature RANGE: 150°F to 2300°F (1650°F op.plts) QUAL. CAT.: 1	JUSTIFICATION: Variable should be added for core cooling function. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 33 FUNCTION: Core Cooling VARIABLE: RCS Loop Flow RANGE: 0 to 110% -12% to 12% flow QUAL. CAT.: 1	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: No core cooling function could be determined or justified for this variable. Deletion is recommended. to D	
TABLE ___ PAGE ___ TABLE 2 PAGE 33 FUNCTION: Core Cooling VARIABLE: Steam Generator Level RANGE: Tube Sheet to Separators QUAL. CAT.: 1	PURPOSE: System Status Information (Type D) FUNCTION: Change to system analysis VARIABLE: RANGE: QUAL. CAT.: 2	JUSTIFICATION: This variable indicates system status rather than core cooling accomplishment. Revise category to 2 to reflect consistency in system status variables. to D	
TABLE ___ PAGE ___ TABLE 2 PAGE 33 FUNCTION: Core Cooling VARIABLE: Condensate Storage Tank Level RANGE: Plant Specific QUAL. CAT.: 1 (3 if not AFW)	PURPOSE: System Status Information (Type D) FUNCTION: Change to system analysis VARIABLE: RANGE: QUAL. CAT.: 2 (3 if not AFW)	JUSTIFICATION: This variable indicates system status rather than core cooling accomplishment. Revise category to 2 to reflect consistency in system status variables. to D	

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TABLE ____ PAGE ____ TABLE 2 PAGE 33 FUNCTION: Core Cooling VARIABLE: Degree of Subcooling RANGE: 200°F subcool to 35°F superheat QUAL. CAT.: 1 (for oper plants 2)	PURPOSE: Analysis of accomplishment of core cooling function (Type B) FUNCTION: Change to system analysis VARIABLE: RANGE: QUAL. CAT.: 3	JUSTIFICATION: This variable is used for analysis purposes; category 3 is consistent with this purpose. No	
TABLE ____ PAGE ____ TABLE 2 PAGE 33 FUNCTION: Maintaining RCS Integrity VARIABLE: RCS Pressure RANGE: 15 psia to 3000 psig (CE 4000) QUAL. CAT.: 1[3]	PURPOSE: Detection, accident mitigation feedback, and validation information (Type B) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ____ PAGE ____ TABLE 2 PAGE 33 FUNCTION: Maintaining RCS Integrity VARIABLE: Pressurizer Level RANGE: Bottom to Top QUAL. CAT.: 1	PURPOSE: System Status Information (Type D) FUNCTION: Change to system analysis VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: This variable indicates system status rather than RCPB integrity function. Category 2 is consistent with system status variables. moved to D	
TABLE ____ PAGE ____ TABLE 2 PAGE 33 FUNCTION: Maintaining RCS Integrity VARIABLE: Primary Sys SRV Positions RANGE: Closed-not closed QUAL. CAT.: 1	PURPOSE: System Status Information (Type D) FUNCTION: Change to system analysis VARIABLE: RANGE: QUAL. CAT.: 3	JUSTIFICATION: Monitoring each RCPB effluent path for valve position is excessive in determining RCPB integrity; SRV position is useful in analysis of loss of integrity. moved to D	
TABLE ____ PAGE ____ TABLE 2 PAGE 33 FUNCTION: Maintaining RCS Integrity VARIABLE: Cont Sump Water Level RANGE: Narrow rge-wide rnge 600,000 gal QUAL. CAT.: 1	PURPOSE: Detection, accident mitigation feedback, and validation information (Type B) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	

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TABLE ___ PAGE ___ TABLE 2 PAGE 33 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, and validation information (Type B) FUNCTION: RCPB Integrity VARIABLE: Containment Pressure RANGE: 0 psig to cont. design pressure QUAL. CAT.: 1	JUSTIFICATION: Variable should be added for RCPB integrity function. Range chosen for this specific function. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 34 FUNCTION: Maintaining Cont. Integrity VARIABLE: Cont Hydrogen Conc. RANGE: 0 to 100% 30% for Ice QUAL. CAT.: 1	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: Function is potential breach of containment for this variable. to C	
TABLE ___ PAGE ___ TABLE 2 PAGE 35 FUNCTION: Fuel Cladding VARIABLE: Core Exit Temp RANGE: 150°F to 2300°F (to 1650°F op plnt) QUAL. CAT.: 1[4]	PURPOSE: Not Determined FUNCTION: Fuel Clad Potential Breach VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: Core exit temperature is designated as Type B variable for core cooling. Validity and sufficiency as a Type C variable to determine potential breach of fuel clad is questioned. Recommend deletion to be consistent with ANS 4.5 No	
TABLE ___ PAGE ___ TABLE 2 PAGE 35 FUNCTION: RC Pressure Boundary VARIABLE: RCS Pressure RANGE: QUAL. CAT.:	PURPOSE: Not Determined FUNCTION: RCPB Potential Breach VARIABLE: Delete variable RANGE: QUAL. CAT.:	JUSTIFICATION: This variable should be verified as the key variable for determining the potential breach of the RCPB. In the interim, recommend deletion to be consistent with ANS 4.5. No	
TABLE ___ PAGE ___ TABLE 2 PAGE 35 FUNCTION: RC Pressure Boundary VARIABLE: Cont. High Range Area Rad RANGE: 1 to 10^7 R/hr QUAL. CAT.: 1[6 10]	PURPOSE: Detection and validation information (Type C) FUNCTION: RCPB Breach VARIABLE: RANGE: 1 to 10^4 R/hr QUAL. CAT.: 3	JUSTIFICATION: Range limit should be lowered to remove excessive conservatism and to be consistent with this function. Validation is the primary purpose; hence, cat. 3 is consistent with this objective. Function is actual RCPB breach. OK	

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TABLE ___ PAGE ___ TABLE 2 PAGE 35 FUNCTION: RC Pressure Boundary VARIABLE: Eff Rad Noble Gas Cond Exh RANGE: 10^{-6} to 10^{5} μCi/cc QUAL. CAT.: 2[9]	PURPOSE: Detection and validation information (Type C) FUNCTION: RCPB Breach VARIABLE: RANGE: 10^{-6} to 10^{-2} μCi/cc QUAL. CAT.: 3	JUSTIFICATION: Primary function is validation of RCPB breach key variable; category 3 and chosen range are consistent with this purpose. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: RC Pressure Boundary VARIABLE: Turb Dr Aux FW Pump Vent RANGE: 10^{-6} to 10^{3} μCi/cc QUAL. CAT.: 2[9]	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: Airborne Rad. Mat. Released VARIABLE: RANGE: 10^{-6} to 10^{-2} μCi/cc QUAL. CAT.: 3	JUSTIFICATION: This variable is a Type E one rather than Type C. Range and category chosen to be consistent with Type E monitors. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: RC Pressure Boundary VARIABLE: Containment Pressure RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, validation, and long term surveillance information (Type C) FUNCTION: RCPB Breach VARIABLE: RANGE: 10 psia to cont. design press. QUAL. CAT.: 1	JUSTIFICATION: Range chosen to be consistent with this function. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: RC Pressure Boundary VARIABLE: Containment Sump Water Level RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, validation, and long term surveillance information (Type C) FUNCTION: VARIABLE: RANGE: Narrow rnge-wide rnge 600,000 gal QUAL. CAT.: 1	JUSTIFICATION: Range chosen to be consistent with this function. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection, accident mitigation feedback, and long-term surveillance information (Type C) FUNCTION: RCPB Breach VARIABLE: RCS Pressure RANGE: 0 to 3000 psig (CE,4000) QUAL. CAT.: 1	JUSTIFICATION: This variable should be added for this function OK	

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TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: Containment VARIABLE: RCS Pressure RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: RANGE: 0 to 3000 psig (CE, 4000) QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: Containment VARIABLE: Containment Pressure RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: RANGE: 10 psia to 3 X D.P., 4 X D.P. QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Potential Breach VARIABLE: Cont. Hydrogen Concentration RANGE: 0 to 10% (30% Ice) QUAL. CAT.: 1	JUSTIFICATION: Function is potential breach of containment for this variable. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 36 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Detection and accident mitigation feedback information (Type C) FUNCTION: Containment Breach VARIABLE: Containment Pressure RANGE: 10 psia to 3 x D.P., 4 X D.P. QUAL. CAT.: 1	JUSTIFICATION: Function is actual breach of containment for this variable as well as potential breach of containment. OK	
TABLE ___ PAGE ___ TABLE 2 PAGE 37 FUNCTION: Secondary Systems VARIABLE: Steam Generator Pressure RANGE: Atmos to 20% above setting QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: C	

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TABLE ___ PAGE ___ TABLE 2 PAGE 37 FUNCTION: Secondary Systems VARIABLE: Aux. Emerg. FW Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2 (1 B&W)	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: Basis for category 1 for B&W plants is questioned. No	
TABLE ___ PAGE ___ TABLE 2 PAGE 37 FUNCTION: Secondary Systems VARIABLE: Main Feedwater Flow RANGE: 0 to 110% design flow QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 37 FUNCTION: Secondary Systems VARIABLE: SRV Positions or Steam Flow RANGE: Closed-not closed QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: (See justification) RANGE: QUAL. CAT.:	JUSTIFICATION: Recommend deletion of code safety valve position requirement. No	
TABLE ___ PAGE ___ TABLE 2 PAGE 37 FUNCTION: Secondary Systems VARIABLE: Eff Rad - Atmos Dump Valve RANGE: 10^1 to 10^3 μCi/cc QUAL. CAT.:	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 37 FUNCTION: Auxiliary Systems VARIABLE: Sump Water Temperature RANGE: 50°F to 250°F QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	AMS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE ____ PAGE ____ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Cont. Atmos. Temperature RANGE: 40°F to 400°F QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE ____ PAGE ____ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Heat Removal by Cont. Coolers RANGE: Plant Specific QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE ____ PAGE ____ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Flow in HPI System RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE ____ PAGE ____ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Flow in LPI System RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	
TABLE ____ PAGE ____ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Emerg. Water Storage Tnk Lvl RANGE: Top to Bottom QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION:	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE ___ PAGE ___ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Accum. Level or Pressure RANGE: 10% to 90% vol, 0 to 750 psi QUAL. CAT.: 2	PURPOSE: System Status Information (Type D) FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Accum. Iso. Valve Positions RANGE: Closed-not closed QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Comp. Cool Water Temperature RANGE: 32°F to 200°F QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 38 FUNCTION: Auxiliary Systems VARIABLE: Component Cool. Water Flow RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 39 FUNCTION: Auxiliary Systems VARIABLE: Letdown Flow - In RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE ___ PAGE ___ TABLE 2 PAGE 39 FUNCTION: Auxiliary Systems VARIABLE: Letdown Flow - Out RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 39 FUNCTION: Auxiliary Systems VARIABLE: Steam Flow to Aux. FW Pumps RANGE: 0 to 110% design flow QUAL. CAT.: 2	PURPOSE: Not Determined FUNCTION: VARIABLE: Delete Variable RANGE: QUAL. CAT.:	JUSTIFICATION: AFW system status is determined by output pump flow; the need for inlet steam flow is tenuous. no	
TABLE ___ PAGE ___ TABLE 2 PAGE 39 FUNCTION: Radwaste Systems VARIABLE: Rad Gas Holdup Tank Press. RANGE: 0 to 150% of design pressure QUAL. CAT.: 3	PURPOSE: Not Determined FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE 2 PAGE 40 FUNCTION: Airborne Rad Mat. Released VARIABLE: Eff Rad Auxiliary Building RANGE: 10^{-6} to 10^{3} μCi/cc QUAL. CAT.: 2[9]	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: VARIABLE: RANGE: 10^{-6} to 10^{2} μCi/cc QUAL. CAT.:	JUSTIFICATION: Upper range value is plant specific; lower values may be acceptable no	
TABLE ___ PAGE ___ TABLE 2 PAGE 40 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: Airborne Rad Mat. Released VARIABLE: Eff. Rad-S/G SR Atmos Dump Vlvs RANGE: (Delete Variable) QUAL. CAT.:	JUSTIFICATION: In general, other measurements provide coverage. Deletion of this variable is recommended. no	

ANS/IEEE INDUSTRY COMMENTS ON NRC REG. GUIDE 1.97 REV. 2 WORKING PAPER F (9-9-80)

NRC POSITION	ANS 4.5 WRITING GROUP COMMENT (9-26-80)		NUPPSCO/NPEC REVIEW COMMENT
TABLE ___ PAGE ___ TABLE 2 PAGE 40 FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: Radioactive effluent assessment by control room operator (Type E) FUNCTION: Airborne Rad. Mat. Released VARIABLE: Eff Rad-Condenser Air Removal RANGE: 10^{-6} to 10^{2} µCi/cc QUAL. CAT.: 2	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	
TABLE ___ PAGE ___ TABLE ___ PAGE ___ FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	PURPOSE: FUNCTION: VARIABLE: RANGE: QUAL. CAT.:	JUSTIFICATION: ✓	

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of METROPOLITAN EDISON COMPANY, et al. (Three Mile Island, Unit 1)	)))))	Docket No. 50-289

AFFIDAVIT OF EDGAR G. HEMMINGER

I, Edgar G. Hemminger, being duly sworn, do depose and state:

1. I am a Mechanical Engineer in the Division of Engineering, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for review and evaluation of structural integrity, operability, and functional capability of safety related mechanical and electrical equipment.

2. I have prepared the statement of Professional Qualifications attached hereto, and, if called upon, would testify as set forth therein.

3. I have answered the UCS September 25, 1980 interrogatory (interrogatories) 2 designated by the initials EGH and the answers given are true and correct to the best of my knowledge.

Edgar G. Hemminger

Edgar G. Hemminger

Subscribed and sworn to
before me this 14th day of
November, 1980

Elizabeth Ann Tipton

Notary Public

My Commission expires: July 1, 1982

EDGAR G. HEMMINGER

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

I am a Mechanical Engineer in the Division of Engineering, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for review and evaluation of the structural integrity, operability, and functional capability of safety related mechanical equipment and components.

I hold a Bachelor of Science Degree in Mechanical Engineering from Ohio University and a Master of Science Degree in Mechanical Engineering from Drexel University and am a licensed Professional Engineer in the State of New York.

From 1965 thru 1979, I was employed by the General Electric Company at the Knolls Atomic Power Laboratory in Schenectady New York. My work experience was in the area of thermal and stress analysis of reactor plant components and equipment. I have specifically evaluated steam generator tube sheets, reactor vessels, nozzles, closure beads and piping systems. Using finite element computer methods I have modeled the vessel closure head and core barrel bolt up region to determine pre load relaxation and lift off for various operating and accident conditions. I have also used results of the above type calculations in conjunction with fracture mechanics methods to establish safe heat up and cooldown pressure and temperature limits for normal plant operation.

In 1973 I completed a one year training program for test and start up of naval reactor plants aboard ship. From 1973 thru 1979 I contributed to the construction, start up and power range physics testing of eight reactor plants aboard ship. My primary duties were to review the test procedures and test data for acceptance testing of naval reactor plants aboard ship and to provide technical support to the shipyard in resolution of equipment problems dealing primarily with valves, pumps, and heat exchangers.

I joined the NRC in October, 1979.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289 (Restart)
(Three Mile Island Nuclear Station, Unit 1)	)	

AFFIDAVIT OF DALE F. THATCHER

Dale F. Thatcher deposes and says under oath as follows:

1. I am a reactor engineer in the Nuclear Regulatory Commission Staff's Instrumentation and Control Systems Branch. I am responsible for the review and evaluation of the instrumentation and control systems of nuclear power generating stations.

2. My professional qualifications are attached.

3. I have answered the UCS September 25, 1980 Interrogatory 6 designated by the initials DFT and the answers given are true and correct to the best of my knowledge.

Dale F. Thatcher
Dale F. Thatcher

Subscribed and sworn to before
me this 14th day of November, 1980.

Elizabeth Ann Tipton
Notary Public

My Commission Expires: July 1, 1982

DALE F. THATCHER
PROFESSIONAL QUALIFICATIONS
INSTRUMENTATION & CONTROL SYSTEMS BRANCH
DIVISION OF SYSTEMS SAFETY

I am a Senior Reactor Engineer in the Instrumentation and Control Systems Branch, Division of Systems Integration, U. S. Nuclear Regulatory Commission.

From May to December 1979, I was assigned to the Bulletins and Orders Task Force as a technical reviewer in the area of instrumentation and control. Just prior to this assignment I was a member of the NRR team which aided in the Three Mile Island Recovery Operation.

In the ICSB, my primary responsibility is to perform technical reviews of the design, fabrication, and operation of instrumentation and control systems for nuclear power plants. This review encompasses evaluation of applicant's safety analysis reports, generic reports and other related information on the instrumentation and control designs.

I graduated from Lehigh University with a Bachelor of Science Degree in Electrical Engineering in June 1971.

From my graduation in June 1971 until my employment at the Commission, I was an Instrumentation Engineer with Gilbert Associates, Inc., an Architect-Engineering company located in Reading, Pennsylvania. My responsibiities included the design and evaluation of various instrumentation and control systems including primarily the areas of reactor protection systems and other safety systems for various domestic nuclear power plants.

I joined the Regulatory staff of the Atomic Energy Commission in March 1974 as a Reactor Engineer. Since the, I have participated in the review of instrumentation control and electrical systems of numerous nuclear power stations and standard plant designs. In addition, I have participated in the formulation of related standards and regulatory guides.

I am a member of the Institute of Electrical and Electronics Engineers (IEEE) and have participated in the development of IEEE Standard 379-1977, "IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class IE Systems" and other proposed standards.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSTION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear Station, Unit 1)	)	
	)	

AFFIDAVIT OF ROBERT G. FITZPATRICK

STATE OF MARYLAND) SS
COUNTY OF MONTGOMERY)

I, Robert G. Fitzpatrick, being duly sworn, depose and state:

1. I am the Section Leader, Electrical Section, of the Power Systems Branch in the Division of Systems Integration, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.

2. I have prepared the statement of Professional Qualifications attached hereto, and, if called upon, would testify as set forth therein.

3. I have answered the UCS September 25, 1980 interrogatories #1 and #3 designated by the initials RGF and the answers given are true and correct to the best of my knowledge.

Robert G. Fitzpatrick

Robert G. Fitzpatrick

Subscribed and sworn to

before me this 14th day of

November, 1980

Elizabeth Ann Fyten

Notary Public

My Commission expires: July 1, 1982

EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
OF ROBERT G. FITZPATRICK

EDUCATION

B.S. Electrical Engineering 1971; Northeastern University, Boston, Mass.

M.S. Electrical Engineering, 1972; Northeastern University, Boston, Mass.

Major: Electrical Power Systems Engineering

PROFESSIONAL QUALIFICATIONS

I am presently Section Leader of the Electrical Section of the Power Systems Branch. In this position, I provide technical supervision and review of the work of reactor systems engineers conducting evaluations of operating reactor problems, license amendments for operating reactors, license applications, generic assessments and special project assignments.

I joined the NRC (AEC) in 1974 as a member of the Electrical, Instrumentation and Controls System Branch and in January 1977 I was assigned to the newly formed Power Systems Branch. My duties during the above periods involved the technical review of electrical systems (onsite and offsite power, and instrumentation and control). For approximately fifteen months following the March 1979 accident at Three Mile Island, I was detailed to the special Three Mile Island Support Group.

From 1972 - 1974 I worked for Yankee Atomic Electric Company in Westboro, Massachusetts. I was assigned to the Electrical and Control Engineering Group and my duties included work on the Yankee operating nuclear plants and the Seabrook Project. (Prior to this I spent 3 years with Yankee as a cooperative education student while attending Northeastern University.)

I am a member of the IEEE and also represent the NRC as a member of IEEE Nuclear Power Engineering Committee Subcommittee 4 "Auxiliary Power Systems." This Committee is charged with developing standards for onsite and offsite power systems.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear Station,	)	
Unit 1)	)	

AFFIDAVIT OF LAURENCE E. PHILLIPS

STATE OF MARYLAND)
COUNTY OF MONTGOMERY) SS

I, Laurence E. Phillips, being duly sworn, depose and state:

1. I am a Section Leader of the Thermal-Hydraulics Section in the Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. I have supervisory responsibility for the review of the reactor core thermal-hydraulic design and behavior including the review of functional requirements for core monitoring systems to provide capability for detection and response to inadequate core cooling conditions.

2. I have prepared the statement of Professional Qualifications attached hereto, and, if called upon, would testify as set forth therein.

3. I have answered the UCS September 25, 1980 interrogatory #4 designated by the initials LEP and the answers given are true and correct to the best of my knowledge.

Laurence E. Phillips

Laurence E. Phillips

Subscribed and sworn to before me
this 14th day of November 1980

Elizabeth Ann Tipton

Notary Public

My Commission expires: July 1, 1982

Laurence E. Phillips

CORE PERFORMANCE BRANCH
DIVISION OF SYSTEMS INTEGRATION
U. S. NUCLEAR REGULATORY COMMISSION

PROFESSIONAL QUALIFICATIONS

I am employed as a Section Leader of the Thermal-Hydraulics Section in the Core Performance Branch of DSI.

I graduated from the University of Cincinnati with a Chemical Engineering degree in 1954. After serving two years as an officer in the United States Army, I have been continuously employed in the nuclear engineering profession since January, 1957. I received a M.S. degree with nuclear physics major from Union College of Schenectady, N. Y., in 1961. I am a registered Professional Engineer, Certificate #E-026547, in the state of Ohio.

In my present work assignment at the NRC, I have supervisory responsibility for the review of the reactor core thermal-hydraulic design submitted in all reactor construction permit and operating license applications. In addition, my section participates in the review of analytical models used in the licensing evaluation of the core thermal-hydraulic behavior under various operating and postulated accident transient conditions. The latter responsibility includes technical review of the functional requirements for core monitoring systems to provide capability for detection and response to inadequate core cooling conditions.

Prior to joining the NRC staff in December, 1974, I was employed by NAI Corporation as a Senior Associate. In this capacity, I was responsible for the development and application of computer codes for analysis of nuclear reactor cores. I acted as a consultant to nuclear operating utilities in the use of these codes for analysis of their operation, and in the solution of general nuclear engineering problems. My tenure at NAI was from 1967 through 1974.

From 1962 to 1967, I was employed by Allis Chalmers Mfg. Co. My assignments during that period included supervisory responsibility for the safety analyses and licensing of the LaCrosse Boiling Water Reactor.

From 1958 to 1962, I was employed by Alco Products where I was project manager for the design, development, and fabrication of heat exchange equipment for nuclear liquid metal projects. Prior to that I was with the Nuclear Division of the Martin Company.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289 (Restart)
	)	
(Three Mile Island Nuclear Station, Unit 1)	)	

AFFIDAVIT OF MEL B. FIELDS

STATE OF MARYLAND)
COUNTY OF MONTGOMERY) SS

I, Mel B. Fields, being duly sworn, depose and state:

1. I am a Containment Systems Engineer in the Containment Systems Branch, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, DC 20555.

2. I have prepared the statement of Professional Qualifications attached hereto, and, if called upon, would testify as set forth therein.

3. I have answered the UCS September 25, 1980 Interrogatory #5 designated by the initials MBF and the answers given are true and correct to the best of my knowledge.

Mel B. Fields

Mel B. Fields

Subscribed and sworn to before me
this 14th day of November 1980

Elizabeth Ann Tipton

Notary Public

My Commission expires: July 1, 1982

Professional Qualifications
Mel B. Fields

I am a Systems Engineer in the Containment Systems Branch of the Office of Nuclear Reactor Regulation. In this position I am responsible for the review and technical evaluation of safety aspects of containment systems.

I graduated from the University of Arizona with a Bachelor of Science Degree in Nuclear Engineering in 1974. I am currently enrolled as a part-time graduate student in the Mechanical Engineering Department of the Catholic University of America in Washington, D. C.

In 1975 I accepted a position as a Reactor Engineer (Intern) in the Containment Systems Branch, Division of Systems Safety, Nuclear Regulatory Commission. My responsibilities included the review and technical evaluation of the safety aspects of containment systems. In this position, I have been responsible for the evaluation of the health and safety aspects related to containment systems for the following nuclear power plants: Black Fox Station, Units Nos. 1 & 2, Grand Gulf Nuclear Station, Units Nos. 1 & 2, North Anna Power Station, Units Nos. 1 & 2, Jamesport Nuclear Station, Units Nos. 1 & 2 and Cherokee and Perkins Nuclear Station, Units Nos. 1, 2 & 3. For the Black Fox Station, I was responsible for reviewing the staff positions and writing the section of the Safety Evaluation Report on the Mark III containment system. In early 1977, I was transferred to another branch, the Power Systems Branch, in the same division where I remained for approximately 1-1/2 years before returning recently to the Containment Systems Branch. I was involved in the preparation of the preliminary clarification of the TMI Action Plan (the September 5, 1980 letter from D. Eisenhut to all licensees and applicants) and the final version of the TMI Action Plan (NUREG-0737) in the areas of the Containment Systems Branch responsibilities.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear Station,	)	
Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S RESPONSES TO UCS INTERROGATORIES OF SEPTEMBER 25, 1980 TO NRC STAFF" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 14th day of November, 1980:

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U.S. Nuclear Regulatory Commission
Washington, DC 20555

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Panel (5)*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Docketing and Service Section (7)*
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U.S. Nuclear Regulatory Commission
Washington, DC 20555

Joseph R. Gray

Joseph R. Gray
Counsel for NRC Staff