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PROPOSED RULE Reg Guide

May 8, 1980



Mr. Ed Wenzinger, Sr.
Reactor Systems - Standards Branch
Office of Standards Development
NRC
Washington, D.C. 20055

Dear Mr. Wenzinger:

RE: Regulatory Guide 1.97

We have enclosed a report on "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environ Conditions During and Following an Accident." This report was prepared by the graduate students listed in partial fulfillment of a course in Nuclear Engineering (NE880.08). The report does not represent opinion of the staff at The Ohio State University. In addition the results and conclusions presented in the report have not undergone the normal internal review process by the faculty or staff at Ohio State.

We hope the information will be of value to you and does represent the technical judgement of the four authors. Please feel free to call if you have any questions.

Sincerely yours,

M. R. Savage
Michael R. Savage
Principal Author

Don W. Miller
Don W. Miller
Faculty Advisor



rlc

cc: D.D. Sharma
C. Jensen
J. Stultz
K. Kiper

Al: Please provide a comparison of their recommendations and our latest draft.
Ed
5/12/80

See conclusions - what set-points do they mean?

A REPORT

PREPARED IN PARTIAL FULFILLMENT OF

NUCLEAR ENGINEERING 880.08

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INTRODUCTION

## INTRODUCTION

The objective of this report is to respond to the proposed revision number two of Regulatory Guide 1.97- Instrumentation for Light-Water-Cooled Nuclear Power Plants to assess plant and environs conditions during and following an accident (See Appendix A). It is our purpose to: (1) Comment on the measured variables listed in the proposed revision, (2) Determine if the ranges given in the proposed revision are appropriate for each measured variable, and (3) Determine if the qualification requirements for each of the instruments used to measure these variables are rigorous enough for a post-accident environment. Inorder to accomplish the above objectives this report will consist of two separate parts. Part 1 will determine which of the measured variables needs to be measured during an accident and the qualification requirements of these instruments. Part 1 will also briefly discuss the ranges of some of the measured variables. Part 2 will analyze the remaining measured variable ranges. Detailed calculations on each of the remaining measured varialbes will be done and from these data (Along with data from other NRC, Industry, Academic, etc. studies) instrument range proposals will be made. The results will be reported in a lated document.

What the study group has accomplished is essentially the first step i.e., to identify the variables that need to be measured, to comment on the ranges of some of the variables, and to discuss the qualification requirements of each instrument. At a later date, the study group will discuss the remaining instrument ranges and the arguments for those ranges.

The work thus far has entailed review of various documents, notably the following:

(1) Report on Monitoring Post-Accident Conditions in Power Reactors Submitted to the U.S. Atomic Energy Commission by Denning, Miller, and Plummer of the Battelle Memorial Institute, (1973)

(2) Functional Requirements for Post-Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generator Station by the writing group ANS 4.5 of the standards committee NUPPSCC (1979)

(3) 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

(4) The final safety analysis reports of the following representative PWR's were also reviewed:

    (i) Davis-Besse

    (ii) Combustion Engineering System 80

    (iii) B & W Standard Plant 241

    (iv) Watts-Bar

(5) TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578 by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, date published July, 1979

(6) TMI-2 Lessons Learned Task Force Final Report by the Office of Nuclear Reactor Regulation, US.

Nuclear Regulatory Commission, Date published: October, 1979

(7) <u>The Draft Standard on Accident Monitoring Instrumentation</u> developed at the August 20-24, 1979 working group meeting held in Pittsburgh, NS-SS-79198

(8) <u>ISA/NPPSC Study Committee on Three Mile Island</u> Minutes for meeting number 79-1, September 10 and 11, 1979

The information obtained from all of these reports is presented in table format for ease of review. An example of the type of table used is shown on the next page. The first four columns in the table are taken directly from the proposed revision to Regulatory guide 1.97. The remaining columns indicate whether the variable is recommended for measurement by the documents listed at each of the column headings. This table is merely a compilation of information from the various documents.

Each variable is separately discussed (by system) in the body of the report. Each discussion is broken down into three separate parts, labeled a, b, and c. Part a includes an argument for or against the need to measure the variable, part b discusses the range of the variable, and part c gives the study group opinion about the qualification type. This section also includes any additional variables needed to monitor a PWR during a post-accident environment.

The end of the report consists of two parts: (1) A complete summary in table form of all the results drawn by the study group and (2) all the conclusions reached by the study group.

SYSTEMS

A.) THE CORE SYSTEM

The core system contains the following measured variables:

(1) Core exist temperature

(2) Control rod position

(3) Neutron flux

Each of the three variables listed above are important in determining the condition of the core in a post-accident nuclear power plant. The information gathered will be used for incore temperature measurements, to provide positive indication that the control rods are fully inserted, and for an indication of approach to critical during start-up (prevent start-up accidents). Therefore, all three should be monitored, The entire core system is listed in the following tables.

Each table contains the following information from the source listed:

| Column Number | Column Name | Source of Information |
|---|---|---|
| 1 | Measured Variable | All this information is taken from: NRC Regulatory Guide 1.97 |
| 2 | Range | Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant |
| 3 | Type | and Environs Conditions During and Following an Accident |
| 4 | Purpose | |
| 5 | SAR Remarks | Taken from the safety analysis reports of the following plants: (i) Davis-Besse (ii) Combustion Engineering System 80 (iii) B&W Standard Plant 241 (iv) Watts-Bar |
| 6 | Battelle BMI-X-647 | Taken from the report on: Monitoring Post-accident Conditions In Power Reactors to the U. S. Atomic Energy Commission, April 9, 1973 |
| 7 | EPRI SR-27 Dec., 1975 | Taken from: Accident Monitoring Instrumentation: Study of the Impact of Proposed Regulatory Guide 1.97 Special report 27 December, 1975 Prepared by: Nuclear Services Corporation 1700 Dell Avenue Campbell, California 95008 Principal investigator: Loren Stanley |
| 8 | NRC Draft RG 1.XX (4-13-74) | Taken From: Unknown |
| 9 | NRC ACRS (Chirman) 5-13-76 | Taken From: Unknown |
| 10 | NRC Lists to Ans. 4.5 7-30-79 | Taken from: The Draft Standard on Accident Monitoring Instrumentation Developed at the August 20-24, 1979 working group meeting held in Pittsburgh |

| Column Number | Column Name | Source of Information |
|---|---|---|
| 11 | ANS 4.5, First Draft 9-7-79 | Taken from: "Instrumentation for Light-Water-Cooled Nuclear Power Plant to Assess Plant and Environs Conditions During and Following an Accident" First Draft, 9-7-79 |

O

# SYSTEM: CORE

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-649 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (SCHIMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| CORE EXIST TEMPERATURE | 150°F TO 2300°F | B,C | ANS-4.5, SECTION 6.2.3 PROVIDE INCORE TEMPERATURE MEASUREMENTS TO IDENTIFY LOCALIZED HOT AREAS. (APPROXIMATELY 50 MEASUREMENTS) | NO MENTION IN PSAR, FSAR'S<br>VERY LARGE RANGE<br>2 THERMOCOUPLES REQUIRED TO COVER THE RANGE 0-530°F 530-2300°F | | | | | ✓ | ✓ 150 TO 2300 |
| CONTROL ROD POSITION | FULL IN OR NOT FULL IN | D | PROVIDE POSITIVE INDICATION THAT THE CONTROL RODS ARE FULLY INSERTED. (MINIMUM 5 DAYS AFTER ACCIDENT) | NOT MENTIONED IN THE PSAR AND FSAR'S<br>BUT AT LEAST FOR B+W PLANTS IN LIMIT AND 0% INDICATION GIVEN IN THE CONTROL ROOM | ✓ | ✓ | ✓ | ✓ | ✓ | ✓ |
| NEUTRON FLUX | 1c/s TO 1% POWER (AT LEAST ONE FISSION COUNTER) | E | ANS-4.5, SECTION 6.2.2 FOR INDICATION OF APPROACH TO CRITICALITY | RANGE ADEQUATELY COVERED IN DAVIS-BESSE FSAR<br>NOT MENTIONED IN PSAR'S | ✓ | ✓ | ✓ | ✓ | ✓ | SHUTDOWN TO 10% [illegible] |

Additional comments and criticism on each of the measured variables are given below:

<u>(1) Core exist temperature</u>

a) This variable is definitely necessary for post-accident monitoring since it will provide in-core temperature measurements to identify localized hot spots present in the core caused by an accident. Thus core damage can be estimated along with the amount of activity released to the primary system.

b) The range seems to be very large and would require at least, two thermocouples per position (0-530°F and 530-2300°F). Using two thermocouples per position, besides being expensive, could be difficult to accomplish. Hence, another method should be employed to obtain the temperature range specified. One possible solution to this problem is to use only one high range thermocouple even though it may have a non-linear operating range. Then by using advanced electronic techniques the non-linear region of this thermocouple response could be linearized, thus extending its range. The range seems to be arbitrary and additional studies are planned in this area.

c) Type B,C would force the nuclear industry to qualify all of the in-core detectors. One possible solution to this problem would be to take approximately one hundred measurements, instead of fifty, but the thermocouples used in these one hundred measurements would be off-the-shelf, instead of class 1E. This would reduce the total cost but still provide the necessary information to monitor hot spots in the core. The actual number of in-core thermocouples used should be examined by further research! A detailed research project in this area should be undertaken to determine which method would lead to the most information for the dollar. Additional work needs to be done in this area.

(2) <u>Control Rod Position</u>

a) There is no question that this variable is necessary for post-accident monitoring. The positive indication that the control rods are fully inserted is absolutly necessary to monitor the plants shutdown condition. With a <u>not-full-in indication</u> the reactor operator would flood the core with a boric acid solution in order to obtain a safe shutdown margin.

b) The range is proper for post-accident monitoring.

c) The type is fine.

(3) <u>Neutron Flux.</u>

a) This measured variable is necessary to monitor and approach to criticality accident that might occur. Thus it should be included in the post-accident instrumentation

b) The low range (1 c/s) implies that a neutron and gamma field will exist within the core. This will restrict the choice of detectors which can be used for obtaining the desired information. Thus additional study should be done in this area at a later date.

c) The type is fine.

## B.) THE REACTOR COOLANT SYSTEM

The reactor coolant system contains the following measured variables:

(1) RCS Hot Leg Temperature

(2) RCS Cold Leg Temperature

(3) RCS Pressure

(4) Pressurizer Level

(5) Degree of Subcooling

(6) Reactor Core Level (added by the study group)

(7) Reactor Coolant Loop Flow

(8) Primary System Safety Relief Valve Position or Flow Through or Pressure in Relief Valve Lines.

(9) Radiation Level in Primary Coolant Water

Each of the nine variables listed above are important in determining the condition of the reactor core cooling system in a post-accident environment. These measured variables supply the reactor operators with the following information: (a) The degree of subcooling present in the primary system, (b) Positive indication of natural circulation in the core, (c) heat balance calculations which are used for direct indication of ECCS operation, (d) Determination of void sizes which might exist in the core, (e) Indication of cooling margin in the core, (f) Insure that the core is covered at all times, (g) Possible primary coolant inventory loss due to a small pipe/valve failure, and (h) Early indication of fuel cladding failure and an estimate of the extent of the damage. Therefore, all nine should be monitored on site. The entire reactor coolant system is listed in the following tables. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This

# SYSTEM: REACTOR COOLANT SYSTEM

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| RCS HOT LEG TEMPERATURE | 150°F TO 750°F | B | ANS-4.5, SECTION 6.2.3 TO AID IN DETERMINING REACTOR SYSTEM SUBCOOLING AND TO PROVIDE INDICATION OF NATURAL CIRCULATION. | SENSOR CAN COVER RANGE, BUT WILL NEED MORE READOUT METERS<br>INDICATOR RANGES<br>520-620°F DAVID-BESSE ±2% FR<br>0-700°F WATTS BAR ±4% FR<br>50-650°F B+W STD PLANTS ±5% FR<br>475-625°F COMB. ENGINEERING STD NO ACC | ✓<br>LISTED TOGETHER WITH THE COLD LEG | ✓<br>LISTED WITH THE COLD LEG | ✓ | ✓ | ✓ | 150°–900°F |
| RCS COLD LEG TEMPERATURE | 150°F TO 750°F | B | ANS-4.5, SECTION 6.2.3 TO PROVIDE INDICATION OF NATURAL CIRCULATION, TO PROVIDE INPUT FOR HEAT BALANCE CALCULATION; FOR DIRECT INDICATION OF ECCS INJECTION | SAME AS ABOVE<br>INDUSTRY AVAILABLE<br>SIGMA 375-675°F ±1% | ✓ | ✓ | ✓ | ✓ | ✓ | 150°–900°F |

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# SYSTEM: REACTOR COOLANT SYSTEM (CONT.)

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 6-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | [illegible] |
|---|---|---|---|---|---|---|---|---|---|---|
| RCS PRESSURE | 15 psia TO 4000 psig | B,C | ANS-4.5, SECTIONS 6.2.3 AND 6.2.4 FOR INDICATION OF AN ACCIDENT AND TO INDICATE THAT ACTIONS MUST BE TAKEN TO MITIGATE AN EVENT. | UPPER LIMIT RANGE SEEMS ARBITRARY SHOULD BE APPROX. EQUAL TO THE BURSTING PRESSURE OF THE RCS. WILL REQUIRE TWO SEPARATE INSTRUMENTS TO COVER RANGE BOURDON TUBE CAN COVER UPPER RANGE 0-3000 psig WATTS BAR, 0-2500 psig DAVIS BESSE FSAR | ✓ | ✓ | ✓ | ✓ | ✓ | 15 psia [illegible] |
| PRESSURIZER LEVEL | BOTTOM TANGENT TO TOP TANGENT | B,D | ANS-4.5, SECTION 6.2.3 LEVEL INDICATION IS REQUIRED TO ASSURE PROPER OPERATION OF THE PRESSURIZER AND TO ASSURE SAFE OPERATION OF HEATERS. IT IS ALSO USED IN CONJUNCTION WITH CHANGES IN REACTOR PRESSURE TO DETERMINE LEAK AND VOID SIZES | CAN BE DONE | ✓ | ✓ | ✓ | ✓ | ✓ | |

# SYSTEM: REACTOR COOLANT SYSTEM (CONT.)

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIRMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 |
|---|---|---|---|---|---|---|---|---|---|---|
| . DEGREE OF SUBCOOLING | 200°F SUBCOOLING TO 35°F SUPERHEAT | E | FOR INDICATION OF MARGIN IN CORE COOLING AND THE NEED FOR EMERGENCY COOLANT ADDITIONS OR REDUCTIONS AS THE MARGIN CHANGES AND TO OBVIATE THE NECESSITY TO CONSULT STEAM TABLES. | CAN BE DONE | | | | | ✓ | 200°F TO 0 |
| REACTOR CORE LEVEL | ENTIRE CORE PLUS SAFETY LEVELS ABOVE AND BELOW THE CORE | TO BE ASSIGNED AT A LATER DATE | TO INSURE THAT THE CORE IS COVERED AT ALL TIMES. | NOT DONE AT THE PRESENT TIME | | ✓ | | | | |
| REACTOR COOLANT LOOP FLOW | 0 TO 120% -20% TO 20% OF DESIGN FLOW | B, D | TO PROVIDE INDICATION THAT THE CORE IS BEING COOLED | 0-120% MOST PROBABLY CAN BE DONE -20% TO 20% NOT MENTIONED IN SARS | | ✓ | ✓ | ✓ | ✓ | 0 TO 120% |

# SYSTEM: REACTOR COOLANT SYSTEM (CONT.)

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIRMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| PRIMARY SYSTEM SAFETY RELIEF VALVE POSITION OR FLOW THROUGH OR PRESSURE IN RELIEF VALVE LINES | CLOSED-NOT CLOSED | B,D | BY THESE MEASUREMENTS THE OPERATOR KNOWS IF THERE IS A PATH OPEN FOR LOSS OF COOLANT AND THAT AN EVENT MAY BE IN PROGRESS. | REQUIRED BY LESSONS LEARNED SHORT TERM RECOMMENDATIONS (2.1.3.a) | | | ✓ | | ✓ | ✓ |
| RADIATION LEVEL IN PRIMARY COOLANT WATER | 10μCi/g TO 10 Ci/g | C | ANS-4.5, SECTION 6.3.2 FOR EARLY INDICATION OF FUEL CLADDING FAILURE AND ESTIMATE OF EXTENT OF DAMAGE. | | | | ✓ | | ✓ | ✓ |

information is given on tables in this section.

Additional comments and criticisms on each of the measured variables are given below:

(1) RCS Hot Leg Temperature

a) This variable is definitly necessary for post-accident monitoring since it aids in determining the reactor subcooling level and it provides indication of natural circulation in the event of RCS pump failure.

b) The range is very large (150°F to 750°F) but present day PWR's (Watts-Bar for example) already have implemented this extended range. Therefore, the study group sees no problem in implementing the range given.

c) The type is fine.

(2) RCS Cold Leg Temperature

a) This variable is definitly necessary for post-accident monitoring since it provides positive indication of natural circulation when the RCS pumps fail and it also provides input data for any heat balance calculations which are used to determine ECCS injection operation.

b) The range should be the same as in the RCS Hot Leg Temperature measured variable. This would simplify construction, procurement, etc. of another type of guage for this system, thus saving the utility money and time. Also the low range will allow accurate calculations of the natural circulation present within the core in the case of pump failure.

c) The type is fine.

(7) RCS Pressure

a) This variable needs to be measured.

b) The range is very large and seems arbitrary. The upper limit range seems arbitrary and it should be equal to approximately two times the design pressure of the reactor coolant system. Again two separate instruments will be required to cover the entire range. Thus, additional research needs to be done in this area at a later date.

c) The type is fine.

(4) Pressurizer Level

(a) This variable needs to be measured during a post-accident condition.

b) The range seems fine but when the water level that is present in the RCS falls below the pressurizer elevation, the primary system water level is completely unknown! Hence, the reactor operator must have additional information to determine the level of water present in the core ( see the reactor core level measured variable).

c) The type is fine.

(5) Degree of Subcooling

a) This variable is not absolutely necessary in a post-accident environment. This measured variable, which cannot be done directly, falls into the "Nice-to-have" category. This variable relieves the reactor operator of having to use steam tables during critical post-accident conditions. This is definitly desirable! But inorder to provide this variable, several primary system variables (RCS temperature and pressure) must be provided and then processed by using advanced electronic techniques (microprocessor). Thus simplicity, relibability, etc. could be lost in providing this variable. Another problem is that it does not consider localized boiling conditions. Even if the degree-of-subcooling meter read a safe condition, boiling still could be occuring deep within the core.

a) (cont.) Operator reliance on this meter thus could lead to a disaster.

b) The range seems arbitrary. Thus additional research in this area will be done at a later date.

c) The type is fine.

(6) Reactor Core Level

a) This measured variable is absolutely necessary for theprotection of the core! With the actual core water level known, no doubt will exist about the core being cooled. But this variable is very difficult and expensive to implement! One possible method would be to include the reactor core level variable and drop several other variables which would no longer be necessary to monitor the core during an accident. Additional research would determine which variables could be dropped. The study group believes that additional study should be done by the NRC,industry, etc. into the possibilities of direct core level measurement in PWR's

b) The range would be plant specific. Aditional research on what safety percentage should be used will be done at a later time.

c) The type letter will be determined at a later time.

(7) Reactor Coolant Loop

a) This variable needs to be measured. But is the actual flow being measured (using pressure devices, etc.) or are the RCS pump speeds used? The actual flow present in the RCS should be used, not the pump speeds. But it could be difficult and expensive to measure.

b) The range is fine with respect to normal flow (0-120%) but the natural circulation limits should be looked into at a later date.

c) The type is fine.

(8) Primary System Safety Relief Valve Position or Flow Through or Pressure in Relief Valve Lines

a) This variable is definitly required for post-accident monitoring. It is required by the TMI-2 lessons learned (short term reccommendations) report. Hence, it must be implemented. But is the actual valve position being measured or is the condition of the electrical, hydralic, Penematic, etc. accuating device being monitored? The actual (mechanical position) condition of the valve is the only information which leaves no doubt as to the status of the valve. But this could be difficult to do.

b) The range is fine.

c) The type is fine.

(9) Radiation Level in Primary Coolant Water

a) This measured variable is required for post-accident monitoring since it provides information about fuel cladding failure and its extent. Should this variable employ sampling at in-frequent intervals or some method in which the RCS could be continuously monitored? This should be looked into.

b) The range seems arbitrary. Thus additional work will be required at a later time.

c) The type is fine.

## C.) CONTAINMENT SYSTEM

The containment system requires the following measured variables:

(1) Containment pressure

(2) Containment atmosphere temperature

(3) Containment hydrogen concentration

(4) Containment isolation valve position

(5) Containment sump water level

(6) High range containment area radiation

The variables to be monitored in the containment are fairly well established and agreed upon. Therefore, each of the six variables listed above are important in determining the condition of the containment during a post accident environment. Since the containment is the last barrier to radiation release, it is vital to monitor its condition during an accident. Thus the instrumentation is generally typed B or C which requires extensive qualifications. These qualifications will assure the public health and safety during an accident. The entire containment system is listed in the following tables. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 27-28 of this report.

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# SYSTEM: CONTAINMENT

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIMMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 [illegible] |
| --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- |
| Containment Pressure | 10 psia pressure to — 3 times design pressure for concrete; 4 times design pressure for steel | B,C | For indication of the integrity of the primary or secondary system pressure boundaries; to indicate the potential for leakage from the containment; to indicate the integrity of the containment. | D-B FSAR 0 to 60 psia with 36 psia design; Foxboro 4 to 85 psia | ✓ | ✓ | ✓ | ✓ | ✓ | ✓ |
| Containment Atmosphere Temperature | 40°F to 400°F | E | For indication of the integrity of the primary or secondary system pressure boundaries, and for indication of the performance of the containment coolant system. | | ✓ | | ✓ | ✓ | ✓ | ✓ |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure) | B,C | For indication of the need and a measure of the performance of the containment hydrogen recombiner. | D-B FSAR 0-5% ± 2% FS | ✓ | | ✓ | ✓ | ✓ | ✓ |

# SYSTEM: CONTAINMENT (CONT)

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE 3NI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (SCHIMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Containment Isolation Valve Position | Closed – not closed | B,C,D | To indicate the status of containment isolation and to provide information on the status of valves in process lines which could carry radioactive materials out of the containment | D-B FSAR can be done | ✓ | | ✓ | | ✓ | ✓ |
| Containment Sump Water Level | Narrow range (sump) Wide range (bottom of containment to 600,000 gallon level equivalent) | B,C | For indication of leakage within the containment and to assure adequate inventory for performance of the ECCS. | sump – yes wider range is possible | ✓ | | ✓ | ✓ | ✓ | ✓ |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr (60keV to 3MeV photons with ±20% accuracy for photons of .1 to 3 MeV) | B,C | To help identify if an accident has degraded beyond calculated values and to indicate its magnitude to determine action to protect public. | none listed in SAR's Kaman 10 to $10^7$ R/hr ±20% General Atomic 1 to $10^8$ R/hr ±20% | ✓ | | ✓ | ✓ | ✓ | ✓ |

Additional comments and criticisms on each of the measured variables are given below:

(1) Containment Pressure

a) This is a variable that should be measured, without question, as indicated by the agreement in the reports (Battelle, EPRI, NRC, ANS, etc.). However, the number and location of the pressure sensors are not listed, except for redundancy.

b) The upper range may be difficult to achieve. The final safety analysis report from Davis-Besse indicated an upper range of about two times design pressure. Hence, additional research into the range will be done at a later time.

c) The variable is typed as B or C which seems quite appropriate.

(2) Containment Atmosphere Temperature

a) This variable is also nearly unanimously agreed upon as one to monitor. At Davis-Besse, containment vessel temperature is measured at 14 points, which probably gives a good indication of the air temperature. The air tends to stratify inside the containment, so a number of monitors would be necessary to give a good indication of the temperature distribution.

b) The range of 40°F to 400°F seems adequate and fairly easy to achieve. But additional study will be conducted at a later time.

c) This variable is typed E, indicating that it supplies no critical information, but does provide information c)(cont) for defense in depht and for diagnosis of the performance of other systems.

(3) Containment Hydrogen Concentration

a) This variable should be measured to indicate the performance of the containment Hydrogen recombiner because of the explosive nature of $H_2$. For containments with external, detachable $H_2$ recombiner, this measurement can indicate when the recombiner whould be attached.

b) The range seems to be somewhat arbitrary- from o to 10% concentration, capable of operating from 10 psia to the maximum design pressure of the containment.

c) Because of this variables' importance in the safety of the public, it is typed B or C.

(4) Containment Isolation Valve Position

a) This is an important variable to be monitored because it is a potential path for radiation release. The agreement among the reports is not as good for this variable. According to the Davis-Besse PSAR, it can be done.

b) The range is fine, no arguments.

c) The type is listed as B,C, or D, indicating some uncertainty as to the degree of qualification. The necessity of recorder for display as required for B and C, is certainly questionable. An indicator would seem to be more appropriate.

(5) Containment Sump Water Level

a) This variable needs to be monitored to provide indication of leakage inside the containment.

b) The range should be plant specific.

c) The type is B,C because of its importance to the safety of the public.

(6) High Range Containment Area Radiation

a) This measurement is agreed upon by the majority of the reports listed. It is important in providing an indication of the severity of any core damage.

b) The range seems to covered with present instrumentation from Kaman and General Atomic, though it may not be qualified for type B or C. Hence, additional work should be done in this area at a later date.

c) The type is B or C and is adequate for this system.

D.) SECONDARY SYSTEM

The secondary system contains the following measured variables:

(1) Steam generator pressure

(2) Steam generator level

(3) Auxiliary Feedwater flow

(4) Main feedwater flow

(5) Safety/Relief valve positions or main steam flow

(6) Radiation in condenser air removal system

(7) Radioactivity in effluent from steam generator safety relief valves or atmospheric dump valves

The following variables definitely needto be monitored in a post-accident environment: a) Steam Generator Pressure, b) Steam Generator Level, c) Auxiliary Feedwater Flow, d) Main Feedwater Flow, and c) Radiation in the Condenser Air Removal System. All of these variables provide information to the reactor operator about the integrity of the secondary system and its ability to remove decay heat from the reactor core. The study group believes that the Safety/Relief valve positions or Main Stream Flow variable does not need to be measured since information about the status of the integrity of the secondary system already exists. Thus this variable can be eliminated from the NRC list. The remaining variable in this list, the radioactivity in the effluent from the steam generator safety relief valves or atmospheric dump valves, would be very difficult to implement and the information it provides is not absolutely necessary during post-accident monitoring. The entire secondary system is listed in the following tables. Each table contains information taken from several reports

# SYSTEM: SECONDARY

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Main Feedwater Flow | 0 to 110% design flow | E | To [illegible] course of water to each steam generator | 0-6.5 x $10^6$ lb/hr | ✓ | | | | ✓ | |
| Safety/Relief Valve Positions or Main Steam Flow | Closed - not closed | B,D | To indicate integrity of Secondary system (vis-a-vis pipe break. | | ✓ | | ✓ | | ✓ | |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^5$ µCi/cc | not listed | To indicate leakage from the primary to the secondary system and measure of noble gas release rate to atmosphere | | ✓ | | ✓ | ✓ | | |
| Radioactivity in Effluent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^5$ µCi/cc | 2,C | As [illegible] [illegible] noble gas release [illegible] to atmosphere | | | | | ✓ | ✓ | ✓ |

# SYSTEM: SECONDARY

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-79 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | D | For indication of integrity of the secondary system, and an indication of capability for decay heat removal | Davis-Besse has a setpoint at 960 psig with an instrument range to 1200 psig or 25% overpressure which satisfies Reg. Guide 1.97 range requirements | ✓ | | ✓ | ✓ | ✓ | ✓ |
| Steam Generator Level | From tube sheet to separators | D | For indication of integrity of the secondary system, and an indication of capability for decay heat removal | Davis-Besse normally mentions range only in terms of inches $H_2O$ [illegible] of [illegible] $H_2O$ [illegible] is needed to evaluate D-B ability to cover required range. | ✓ | | ✓ | ✓ | ✓ | ✓ |
| Auxiliary Feedwater Flow | at 110% design flow | D | To indicate an adequate source of water to each steam generator | D-B range 0-100% ± 2% | | ✓ | ✓ | | ✓ | ✓ |

and/ or standards which deal with post-accident monitoring. This information is given in the tables on pages 8 AND 9 of this report.

Additional comments and criticisms on each of the measured variables are given below:

(1) Steam Generator Pressure

a) This variable should be measured since it provides the operator with information about the integrity of the secondary system and its ability to remove decay heat from the core. The steam generator pressure can also be used to indicate a steam line break. Thus the steam generator pressure is an important variable in the initial phases of an accident to help deduce the nature of the accident.

b) The range should be specified so that the entire range of the steam generator is considered. The range gived in the NRC report is not very clear. Additional research should be done in this area and the resutls will be reported at a later date in a follow-up report.

c) The type is fine.

(2) Steam Generator Level

a) The steam generator level is an important link in the heat transfer path to the ultimate heat sink. Thus, this variable should be monitored to indicate a loss of this heat transfer path.

b) The range is fine for dual-pass steam generator but what about single-pass steam generators? This range should be plant specific.

c) The type is fine.

(3) Auxiliary Feedback Flow

a) NUREG-0578, because of the accident at Three Mile Island, requires that this variable be monitored with safety-grade equipment. This variable must be displayed in the control room and powered from the emergency bus. Thus, this variable must be monitored during a post-accident condition.

b) The range is fine.

c) The type is fine.

(4) Main feedwater Flow

a) The main feedwater flow is definitely needed during any type of post-accident environment. It provides information about an adequate source of water to each steam generator which is used to remove decay heat from the core during an accident. Thus, the main feedwater flow should be measured and is already monitored on present-day PWR's.

b) The range is fine.

c) Theauxiliary and main feedwater flow should have the same type! It does not seem logical to type one D while the other E. The study group believes that the main feedwater flow should be re-typed D.

(5) Safety/ Relief Valve Positions or Main stream Flow

a) The study group believes that this variable is not needed for the monitoring of an accident. If one of the main stream safety valve opens, the noise produced by the steam escaping would be enough to indicate to the reactor operator that there was a major problem. The usefulness of this variable during a small steam leak is questioned. The steam generated pressure is a much better indication of a small secondary system pipe break.

b) The range is fine if the NRC still believes that the information provided by this variable is absolutely necessary.

c) The type should be D or even E. The study group believes the NRC is over-compensating because of the accident at Three Mile Island. A small leak in one of the safety relief valves would cause, the steam generator pressure and level to drop, thus giving the operator the information necessary to take corrective action. If the NRC still believes this measured variable to be necessary, the type could be changed to type E, thus reducing the cost of implementing this system on all of the safety relief valves (The number of safety relief valves is large and it depends upon the plant).

(6) <u>Radiation in Condenser Air Removal System</u>

a) This variable is necessary during an accident to indicate to the reactor operator the leakage of noble gases from the primary to the secondary system. This information is useful for accident analysis. Any increase in the radiation level indicates a steam generator tube rupture.

b) The range seems high. Two or three orders of magnitudes seems appropriate. Additional research in this area should be done and reported in a follow-up report at a later date.

c) The NRC did not assign a type to this variable. Therefore, the study group assigned a type C to this variable since it indicates a breach of one of the fission product barriers.

(7) <u>Radioactivity in Effluent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves</u>

a) This variable would be difficult to implement. One possible method consists of monitoring the radiation in the main line. Two monitors, one above the relief valves and the other downstream of the relief valves, could be used. This method would greatly reduce the number of monitors required (since the number of atmospheric dump valves is large).

b) The range see s high. Two or three orders of magnitudes seems appropriate. Additional research in this area should be done and reported in a follow-up report at a later date.

c) The type should be type C.

F.) AUXILIARY SYSTEM

The auxiliary system contains the following measured variables:

(1) Containment Spray Flow

(2) Flow in High Pressure Injection System

(3) Flow in Low Pressure Injection System

(4) Emergency Coolant Water Storage Tank Level

(5) Pressure of $N_2$ in the Accumulator Tank (added)

(6) Accumulator Tank Level

(7) Accumulator Isolation Valve Positions

(8) RHR System Flow

(9) RHR Heat Exchanger Out Temperature

(10) Component Cooling Water Temperature

(11) Component Cooling Water Flow

(12) Flow in the Ultimate Heat Sink (UHS) Loop

(13) Temperature in Ultimate Heat Sink Loop

(14) Ultimate Heat Sink Level

(15) Heat Removal by the Containment Fan Coolers

(16) Boric Acid Charging Flow

(17) Letdown Flow

(18) Sump Level in Spaces of Equipment Required for Safety

Comments on each follow. After the comments are listings of data collected.

# SYSTEM: AUXILARY

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Containment Spray Flow | 0 to 110% Design Flow | D | For Indication of System Operation | | | X | | X | X | X |
| Flow in HPI System | 0 to 110% Design Flow | D | For indication of system operation | | | X | | | | X |
| Flow in LPI System | 0 to 110% design flow | D | For indication of system operation | | | X | | | | X |
| Emergency Coolant Water Storage Tank Level | Top to bottom | D | To determine the amount of water discharged by the ECCS. This provides indication of the nature of the accident, indication of the performance of the ECCS, and indication of the necessity for operator action. | | X | | X | X | X | X |
| Accumulator Tank level | Top to bottom | D | To indicate whether the tanks have injected to the reactor coolant system. | | | | | | X | X |
| Accumulator isolation valve positions | Closed-not closed | D | To indicate state of the isolation valves. (Per Regulatory Guide 1.47) | | | | | | | X |
| RHR System Flow | 0 to 110% design flow | D | For indication of system operation. | | | X | | X | X | X |
| RHR Heat Exchanger Out Temperature | 32 F to 350 F | D | For indication of system operation. | | | | | | | 32-150 |

# SYSTEM: AUXILARY H

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 |
|---|---|---|---|---|---|---|---|---|---|---|
| Component Cooling Water Temperature | 32 F to 200 F | D | For indication of system operation. | | | | | | X | 32 15' |
| Component Cooling Water Flow | 0 to 110% design flow | D | For indication of system operation. | | | | | | X | X |
| Flow in UHS Loop | 0 to 110% design flow | D | For indication of system operation. | | | | | | X | X |
| Temperature in Ultimate Heat Sink Loop | 30 F to 150 F | D | For indication of system operation. | | | | | X | X | 30 13' |
| Ultimate Heat Sink Level | Plant Specific | D | To insure adequate source of cooling water. | | | | | X | X | X |
| Heat Removal by the Containment Fan Coolers | Plant specific | B | To indicate system operation | | | | | | | X |
| Boric Acid Charging Flow | 0 to 110% design flow | B | To provide indication of reactor cooling and inventory control and maintain adequate concentration for shutdown margin. | | | X | | | X | X |
| Letdown Flow | 0 to 110% design flow | D | For indication of reactor coolant inventory control and boron concentration control. | | | | | | X | X |

# SYSTEM: AUXILARY

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | D | To monitor environmental conditions of equipment in closed spaces. | | | | | X | | X |

(1) Containment Spray Flow

a) The containment spray flow measurement is needed to indicate system operation, and therefore should be included in a list to be measured. It is most likely that such instrumentation already exists.

b) The range requirement came under some question as to the necessity for 0-110% design flow when to the best of our knowledge the system was either on or off---no throttled position was possible. Using the flow indication for a measure of water in containment would be redundent, and could be measured more directly by sump level.

c) The classification type D, for indication of performance of individual safety systems, seems appropriate. The qualification requirements have probably been met on indicator already installed.

(2) Flow in HPI System

a) Flow measurement in HPI system is required to indicate safety system operation. Instrumentation already exists. (B & W)

b) The range requirement of 0-110% design flow is not disputed because the HPI system can be throttled.

c) Instrument type is okay.

(3) Flow in LPI System

a) Flow measurement in LPI system is required to indicate safety system operation. Instrumentation most probably installed.

b) The range requirement of 0-110% design flow is acceptable because the LPI system flow can be throttled.

c) Type of qualification necessary is acceptable.

(4) Emergency Coolant Water Storage Tank Level

a) There is little question in the group concensus or among the studies reviewed that this variable should be measured.

b) The range as listed in Reg. Guide 1.97 Rev. 2 (proposed) is ambiguous and should be revised to indicate say from 1 foot from top at tank center to switchover setpoint to auxiliary input as the bottom level. The bottom of tank level indication below this point is useless because the tank is no longer used for a water supply. As for the purpose, the group feels that the NRC may be overstating the amount of information that can be reasonably infered from this one measurement when saying that the level indication would provide "..indication of the nature of the accident... of the performance of the ECCS.. of the necessity for operator action..." The NRC indicated that they want (require) the most direct measurement means, and yet they state that they want the level indication to provide information about things that can be determined more directly.

(5) Pressure of $N_2$ in the Accumulator Tank (added)

a) This variable was added to the list of variables required for completeness because in normal measurement methods the blanket pressure must be known so that the water level can be determined. $N_2$ pressure is already measured.

b) Range should be from 0-110% design pressure or perhaps from 0-3000 psia to indicate failure of the accumulator isolation valves.

c) The type should be D as the level indicator is required to be.

(6) Accumulator Tank Level

a) Variable should be measured and is measured per B & W TMI (Davis-Besse?)

b) The range perhaps should be from the top of the gas blanket (when in system stand by, that is not activated) to the top of the level indicator sensor which is say within 1 foot of tank bottom. This would pin valves down more reasonably that saying range is from "top to bottom".

c) Qualification type D is acceptable.

(7) Accumulator Isolation Valve Positions

a) Valve position is required to be measured in Reg. Guide 1.97 so the positions are indicated already.

b) Range is logical and acceptable.

c) Qualification requirements are acceptable.

(8) RHR System Flow

a) There is a concensus among the group and the reports studied that this variable should be measured. There is indication that this variable is already measured.

b) The range is acceptable.

c) The qualification type is reasonable.

(9) RHR Heat Exchanger Out Temperature.

a) It is agreed that this variable should be measured, but some confusion exists as to which outlet stream should be measured, or perhaps both.

b) The range analysis has been postponed for the present.

c) The qualification requirements are reasonable.

(10) Component Cooling Water Temperature

a) The water temperature should be measured, but some question exists as to whether the inlet or outlet or both temperatures are intended to be measured. There is additionally the question of whether the bulk or average temperature should be taken to indicate cooling of safety components. Cooling water temperature is currently being monitored.

b) The applicability of the range has been left for further study at a future date.

c) The qualification requirements are reasonable

(11) Component Cooling Water Flow

a) This variable needs to be measured and is measured in existing plants.

b) Range is acceptable

c) Qualification type is acceptable.

(12) Flow In UHS Loop

a) This variable needs to be and is measured in most plants.

b) Range is acceptable.

c) Qualification type is acceptable.

(13) Temperature in Ultimate Heat Sink Loop

a) The temperature should be measured, but the temperature to be measured is not clearly defined. Average inlet and outlet temperatures have been suggested for measurement.

b) The range analysis has been defered until a later date.

c) The qualification tyre is acceptable.

(14) Ultimate Heat Sink Level

a) The level should be measured, and is measured as a normal plant variable. There is some confusion as to what constitutes the ultimate heat sink. This group assumed that the heat sink was the water cooling the condenser and going through the cooling tower or rejected to a body of water.

b) The range, althoulgh plant specific, needs more definition such as general guidelines should be established. it may be possible to relate the range to some plant variable such as power level.

c) The qualification tyre is acceptable.

(15) Heat Removal By The Containment Fan Coolers

a) The heat removal rate was only mentioned by the ANS Committee 4.5 as an indication of operation of the containment air coolers. There is some doubt about the necessity to measure the heat rate when fhe fan currents can be used to indicate system operation perhaps with component cooling water flow indication. If the heat removal rate must be used, the inidcation can be determined from the inlet and outlet temperature and the component cooling water flow rate through the coolers.

b) The range is plant specific, but as mentioned before there should be more information given to establish general guidelines. It may be possible to correlate a general range with initial load parameters and burnup and EFPD's.

c) The qualification requirements proposed may make the equipement very difficult and expensive to implement.

(16) Boric Acid Charging Flow

a) There is strong evidence to indicate that this variable should be measured, and that is measured during the course of normal plant operations. There is serious doubt concerning several items mentioned in the purpose. There is much doubt that this flow alone can provide sufficient flow to insure core cooling, and that other information is required before the remainder of the items mentioned in the purpose can be alone. It seems clear that the NRC has overstated the amount of information available from this one flow measurement.

b) The range is reasonable because the flow can be throttled.

c) The qualification requirements appear to be excessive. it would be logical to assume that the charging flow instrumentation would be no more important than HPI or LPI flow and accordingly the requirements should be dropped to D.

(17) Letdown Flow

a) This variable should be measured, and is most probably measured in the course of normal plant operation. In the purpose section, it appears that the NRC can be determine the reactor coolant inventory and boron concentration from the letdown flow measurement. I believe that the flow measurement is important in determining the listed information.

b) The range on the flow is acceptable because the flow can be varied.

c) The qualification requirements are satisfactory.

(18) Sump Level is Spaces of Equipement Required for Safety

a) There is some question as to the value of this variable as it may be possible to render the equipement inoperable from the containment spray or other spray from ruptured pipes without the sump level indicators registering danger levels. A large amount of development time and cost would be required to implement, and the benefit may not be worth the cost. Again the NRC overstates in the purpose suggested rewording would be to monitor an environmental condition for potential of equipment failure.

b) Range is acceptable except that for multiple equipment and multiple failure levels; only the lowest need be monitored.perhaps is remedial action is available.

c) The qualification requirements are acceptable.

F.) RADWASTE SYSTEM

The radwaste system contains the following measured variables:

(1) High Level Radioactive Liquid Tank Level

(2) Radioactive Gas Hold-up Tank Pressure

Each of the two variables listed above provide the reactor operator with information-in-depth about the radioactive waste system. Although not absolutely necessary for post-accident monitoring, these two systems provide information about the available volume to store the primary coolant and the capacity to store any waste gases which buildup in the primary system. The entire radwaste system is listed in the following tables. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 8+9 of this report.

Additional comments and criticisms on each of the measured variables are given below:

<u>(1) High Level Radioactive Liquid Tank Level</u>

a) During normal operation the liquid waste tank stores radioactive liquids.

(1) To allow time for decay

(2) To provide temporary holdup until they can be processed by the liquid waste system treatment system.

At any moment of time during normal operation or after an accident it is important to know:

(1) The available capacity to store more liquid.

(2) The occurrence of a leak (estimated probability $\underline{10^{-2}}$)

# SYSTEM: RADWASTE SYSTEM

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIRMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| High Level Radioactivity Liquid Tank Level | Top to Bottom | E | Available volume to store primary coolant | | – | – | – | – | ✓ | ✓ |
| Radioactive Gas Holdup Tank Pressure | 0 to 150% design pressure | E | Available capacity to store waste gases | | – | – | – | – | – | ✓ |

Measuring liquid level is the simplest method of accomplishing both of these purposes.

One main problem that the study group discussed about this measured variable is: how much of the primary coolant can the high level radioactive liquid tank store? All of the primary water inventory? Part of it? This is not made clear by the NRC reports. The ideal condition would by to have storage capacity amounting to 2 or 3 times the entire inventory! Thus during an accident any radioactive water could be stored in tanks not on the containment floor as in the Three-Mile Island accident. Hence, clean-up would be fairly easy. The NRC should be more precise in this area.

b) The range should be plant specific not just top to bottom. Many utilities might use only indicator lights instead of meters. Hence, the range should read: Plant specific, meters only.

c) The study group, with one exception, believes that the instrumentation for this purpose should be qualified Type E, which concurs with the opinion of the NRC. However, one member favors a better qualification because of the following argument: In the case of an accident, enormous amounts of water will be needed for cooling, flushing, and cleaning. This volume of water will far exceed the normal primary coolant volume. Therefore, information on the availability of storage water will be a must in such a case. Thus a superior qualification possibly Type C or D, should be used.

(2) Radioactive Gas Holdup Tank Pressure

a) The waste gas storage tank is used to store radioactive gases which have been processed through the waste gas treatment system. The storage of this gas permits time for radioactive decay to occur. Thus any releases to the atmosphere will contain far less activity. Measurement of the pressure within this tank is needed to indicate:

1) The available capacity

2) Any inadevertent release of gases.

the probability of an inadvertent release is $\frac{10^{-2}}{yr}$.

The maximum inventory is 9.5 x $10^4$ curies.

b) The range should extend to the bursting pressure of the tank. Thus the range should be plant specific.

c) The same reasons that were given for the previous variable also apply. The type should be upgraded to type C or D.

G.) VENTILATION SYSTEM

The ventilation system contains the following measured variables:

(1) Emergency Ventilation Damper Position

(2) Temperature of Space in Vicinity of Equipment Required for Safety

Each of the two variables listed above are already being monitored in the present day PWR plants. These two systems are not absolutely necessary for post-accident monitoring.but they do provide the control room operator with information about the performance of individual safety systems. The ventilation system is listed in the following table. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 8+9 of this report.

Additional comments and criticisms on each of the measured variables are given below:

(1) Emergency Ventilation Damper Position

a) This variable is already being monitored in present day PWR's.

b) The range is obvious.

c) The type could force the upgrading of the measurement of this variable but the cost will be minor and should not be dificult to implement.

(2) Temperature of Space in Vicinity of Equipment Required for Safety

a) This variable should be monitored during an accident since any high temperature near vital safety equipment could jeopardize the safety of the plant and the public. Also and elevated temperatures near vital equipment could indicate either inadequate ventilation or faulty equipment. This is a useful variable but the number of monitors should be limited.

b) The range seems arbitrary and depends upon many factors. These factors range from the actual construction of the plant to the location of the plant in the United States. This variable should be plant specific. Additional work will be done at a later time to clarify this area.

c) The type is fine.

# SYSTEM: VENTILATION

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIRMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 |
|---|---|---|---|---|---|---|---|---|---|---|
| Emergency Ventilation Damper Position | open-closed Status | D | To ensure proper ventilation under accident conditions | already done | | | | ✓ | ✓ | ✓ |
| Temperature of Space in Vicinity of Equipment Required for Safety | 30°F to 180°F | D | To monitor environmental conditions of equipment in closed spaces | already done | ✓ | | ✓ | ✓ | ✓ | ✓ |
| | | | | | | | | | | |

E.) POWER SUPPLY SYSTEM

The power supply system contains the following measured variables:

(1) Status of Class IE Power Supplies and Systems

(2) Status of Non-Class IE Power Supplies and Systems

What exactly does "status" mean? Listed below are a few possible methods which could be used to monitor the status of and power supply:

a) Each power supply could have its own separate analog or digital meter to indicate its status.

b) Each power supply could be assigned a range of acceptable voltages and currents and then by using analog or digital techniques two on/off (acceptable voltage or current/non-acceptable voltage or current) indicators could be read-out in the control room. Two would be for acceptable/non-acceptable voltages while the other two would be for acceptable/non-acceptable current levels.

c) Simple lights could be used to indicate whether the power supply was on or off.

d) etc.

Thus many possible methods can be devised to monitor the "status" of a power supply but most of these methods would be very expensive. Hence, the question must be asked: Is the information worth the cost?

The status of Class IE power supplies and systems are already monitored as part of the safety systems. This is to insure that in case of an accident, electric power will always be available to insure the safety of the plant(and hence the public). Thus

# SYSTEM: POWER SUPPLIES

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BAI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 ([illegible]) | NRC LISTS TO ANS4.5 7-20-79 | ANS 4.5 [illegible] |
|---|---|---|---|---|---|---|---|---|---|---|
| STATUS OF CLASS 1E POWER SUPPLIES AND SYSTEMS | VOLTAGES AND CURRENTS | D | TO ENSURE AN ADEQUATE SOURCE OF ELECTRIC POWER FOR SAFETY SYSTEMS. | | ✓ | | ✓ | ✓ | ✓ | ✓ |
| STATUS OF NON-CLASS 1E POWER SUPPLIES AND SYSTEMS | VOLTAGES AND CURRENTS | E | TO INDICATE AN ADEQUATE SOURCE OF ELECTRIC POWER | | | | ✓ | ✓ | ✓ | ✓ |

the status of class 1E power supplies and systems are already being monitored.

The entire power supply system, as used in post-accident monitoring, is listed in the following table.

Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 8-9 of this report.

Additional comments and criticisms on each of the measured variables are given below:

(1) <u>Status of Class 1E Power Supplies and Systems</u>

a) This is already being done, to some extent, as part of the safety systems. Is further redundancy desirable and/or necessary? If new systems are added which are especially built for post-accident monitoring, then class 1E standards must be met, thus increasing the cost of a nuclear power plant. If individual meters are employed to monitor the status of power supplies then the reactor operator must know: What is an acceptable voltage or current? Here again, individual judgement and knowledge will determine what is acceptable. Thus the operator must be more knowledgable and in a crises (as in Three-Mile Island) mistakes could be made. Thus additional research needs to be done in this area to determine the best possible approach. This research should precede as follows: (1) Check into the monitoring of class 1E power supplies in several safety systems used in present day PWR's, (2) Determine what is meant by status, (3) Determine the cost of using individual meters versus advanced electronic means and (4) Look into the human factor.

b) The range is fine, excepr for the fact that the method of measuring the voltage and current, as outlined above, must be determined.

c) The type is fine.

(2) Status of Non-Class 1E Power Supplies and Systems

a) This measured variable is questionable. Present plant design already includes two or three levels of redundancy in this area. Adding this to present or new plants would greatly increase the cost of these plants without increasing their safety.

b) The range is fine, except again the method of measuring the voltages and currents will determine the complexity of implementing these measurments.

c) The type is fine.

d) This variable could be dropped off the list of variables monitored in a post-accident environment.

I.) RADIATION EXPOSURE RATES INSIDE BUILDING OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT SYSTEM

The radiation exposure rates inside building or areas where access is required to service safety related equipment system contains the following measured variables:

(1) Radiation Exposure Rates

This variable should be measured since it provides high-range radiation exposure rates at various locations throughout the plant. The entire system is listed in the following tables. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 8-9 of this report.

<u>(1) Radiation Exposure Rates</u>

a) The entire study group believes that this variable should be measured. This is being done at present.

b) The range is questionable. Several questions must be answered before any conclusions can be made. A few examples are given below:

1) Is the upper range ($10^4$ R/hr) justified?

2) Is this measurement capability required in excess to already existing facilities?

3) etc.

Further work will be done at a later date in this area.

c) The type is fine.

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SYSTEM: Radiation exposure rates inside buildings or areas where access is required to service safety related equipment

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-79 | NRC ACRS 8-13-76 (CHIRMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Radiation Exposure Rate | VARIES | E | For measurement of high-range radiation exposure rates at various locations | KAMAN SCIENCE MOD KHRA-5 | – | – | ✓ | ✓ | ✓ | ✓ |

## J.) AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT

The airborne radioactive materials released from the plant system contains the following measured variables:

(1) Effluent Radioactivity-Noble Gases

(2) Effluent Radioactivity-High Range Radiohalogens and Particulates

(3) Environs Radioactive-High Range Exposure Rate

(4) Environs Radioactivity-Radiohalogens and Particulates

(5) Plant and Environs Radioactivity (Portable Instruments)

Each of the five variables listed above are important in determining the radiation field which exists in and around a post-accident power plant. Therefore, all five should be monitored. The entire core system is listed in the following tables. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 8-9 of this report.

Additional comments and criticisms on each of the measured variables are given below:

<u>(1) Effluent Radioactivity-Noble Gases</u>

a)This variable is definitly needed during an accident since it provides the operator with information about the release of radioactive noble gases on a continuous basis. But the variable description is not specific enough and several questions arise which must be answered if implementation in present-day PWR's is to proceed efficiently. Some of these questions are listed below:

(i) How many detectors should be used?

(ii) Where should these detectors be placed?

# SYSTEM: AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| ENVIRONS Radioactivity - High Range Exposure Rate | | E | for estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability, approx. 16 to 20 locations - site dependent | VICTOREEN 845 $10^{4}$ to $10^{4}$ R/hr | — | — | — | — | ✓ | ✓ |
| ENVIRONS Radioactivity - Radioiodine and Particulates | | E | same as above. [illegible] undetermined sediment by continuous collection capability | KAMAN KPIM $10^{-9}$-$10^{-3}$ $\mu$ci/cc | — | — | — | — | ✓ | ✓ |

# SYSTEM: AIRBORNE RADIOACTIVE MATERIAL RELEASED FROM THE PLANT

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIRMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Effluent Radioactivity - High Range Radiohalogens and Particulates<br>· Unheated -<br>· HPEA filter, min of 2" of TEDA -<br>· HPEA filter, min of 4" of TEDA - | | E | To provide the operator with information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by monitoring (measurements) of samples for radiohalogens and for particulates. | KAMAN SCIENCE KDA-1 (photons) $10^{-3}-10^{+2}$ R/hr | – | – | | | | ✓ |

# SYSTEM: Airborne Radioactive Materials Released from the Plant

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE 3MI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Plant and Environs Radioactivity (Portable instrument) | | E | During and following accident, to monitor radiation and airborne radioactivity conc. in many areas throughout the facility where it is impractical to install stationary monitors capable of covering normal and accident levels | KAMAN SCIENCE CORPORATION | - | - | - | - | - | ✓ |

# SYSTEM: AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNWL-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS4.5 7-30-79 | ANS 4.5 |
|---|---|---|---|---|---|---|---|---|---|---|
| Effluent Radioactivity – Noble Gases | | | ANS 4.5 Section 6.2.6 | | | | | | | |
| . containment | $10^{-7}$ to $10^{5}$ µci/cc (Xe-133 calibration) | | To provide operator with information regarding release of radioactive noble gases on a continuous basis | KAMAN SCIENCE Corporation | | | | | | |
| . Secondary containment | $10^{-7}$ to $10^{4}$ µci/cc (Xe-133 calibration) | | | $5\times10^{-7}$ to $10^{5}$ µci/cc | – | – | ✓ | ✓ | ✓ | ✓ |
| . Aux. Building (including buildings containing primary system gases, e.g. Waste gas decay tank) | $10^{-7}$ to $10^{3}$ µci/cc | E | | KHRG (Mid 1980) | | | | | | |
| . Other Release Points (including fuel handling area if separate from auxiliary building) | $10^{-7}$ to $10^{2}$ µci/cc | | | | | | | | | |

(iii) Should many detectors in many places be used or should a single detector in the optimium place be used?

(iv) Do you need a local readout or a remote indicator in the control room?

Therefore, the information provided by the measurement of this variable is helpful in post-accident monitoring but the requirements and specifications on the instrumentation needed should be made more specific.

b) Many separate ranges will be needed to cover the many ranges given in the NRC standard. This could add sequently to the cost of the power plant. Additional research into the ranges will be conducted at a later time and reported in a follow-up report.

c) The type is fine

(2) Effluent Radioactivity-High Range Radiohalogens and Particulates

a) This variable should be measured since it provides the operator with information about the release of radioactive halogens and particulates. Again the variable description is not specific enough and before implementation can occur, the same questions must be answered. See measured variables: Effluent Radioactivity-Noble Gases section 1 a.

b) At least three separate meters, possibly many more, will be needed to cover the ranges as given by the NRC standard. Again expense could be high! Additional research into the ranges will be conducted at a later time and reported in a follow-up report.

c) The type is fine.

(3) Environs Radioactivity-High Range Exposure Rate

a) This variable should be measured since it provides information necessary to estimate the release rates of radioactive materials during an accident. Again the variable description is not specific enough and before proper implementation can occur, the same questions should be answered. See measured vairable: Effluent Radioactivity-Noble Gases section 1a. One additional question can be asked: Should all of the information from the detectors used to monitor this variable go directly to the main control room or should a special emergency action room be used? The special emergency action room could be run by civil defense, public health, and/or NRC officials. This would leave the reactor operators free to bring the reactor under control and eventually to a safe cold-shutdown condition.

b) Againtthe total number of meters required to implement the ranges, as given, could be high. Thus the cost could be high. Additional research into the ranges will be conducted at a later time and the results will be reported in a follow-up report.

c) The type is fine.

(4) Environs Radioactivity-Radiohalogens and Particulates

a) This variable should be measured since it provides information necessary to estimate the release rates of radioactive materials during an accident. Again the variable description is not specific enough and before proper implementation can occur, the same questions as outlined in section 1a should be answered. See measured variable: Effluent Radioactivity-Noble Gases section 1 a. This information should go to the emergency action room instead of the main control panel.

b) Again the total number of meters required to implement the ranges, as given in the NRC report, could be high. Thus additional research into the ranges and the methods used to detect them, will be conducted at a later time and the results will be reported in a follow-up report.

c) The type is fine.

(5) Plant and Environs Radioactivity (Portable Instruments)

a) This variable should be measured since it provides information about airborne radioactivity concentrations in areas of the facility where it is impractical to install stationary monitors capable of covering both normal and accident radioactivity levels. Again the variable description is not specific enough and before proper implementation can occur, the questions asked in section 1a need to be answered. This information should also go to the emergency action room instead of the main control panel.

b) Again the total number of meters required to implement the ranges, as given in the NRC report, could be high. Thus, additional research into the proper detectors and their ranges, will be conducted at a later time and the results will be reported in a follow-up report.

c) The type is fine.

K.) POST-ACCIDENT SAMPLING CAPABILITY SYSTEM

L.) POST-ACCIDENT ANALYSIS CAPABILITY (ON-SITE) SYSTEM

The post-accident sampling capability system contains the following variables:

(1) Primary Coolant Sumps

(2) Containment Air

The post-accident analysis capability (on-site) system contains the following measured variables:

(1) Gamma-Ray Spectrum

(2) pH

(3) Hydrogen

(4) Oxygen

(5) Boron

Post-Accident Sampling and Analysis

(1) Sampling Capability Primary Coolant Sumps and Containment Air

a) These two media should be sampled.

b) Range analysis has been defered until later.

c) Type is N/A.

(2) Analysis Capability (Onsite)

a) The analysis capabilities listed should be monitored and are presently being used during normal mode plant operation. It is logical to assume that the majority of the equipment exists, therefore additional equipment and/or personnel may not be necessary.

b) The range should be developed more before comments are feasible.

c) Type is N/A.

# SYSTEM: POST-ACCIDENT ANALYSIS

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BMI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| Post-Accident Sampling Capability<br>Primary Coolant<br>Sumps<br>Containment Air | As required based on Reg Guide 1.4 guidelines | N/A | Provide means for safe and convient sampling. These provisions should include:<br>1. Shielding to maintain radiation doses ALARA,<br>2. sample containers with container-sampling port connections capability,<br>3. Capability of sampling under under primary system pressure and negative pressure.<br>4. handling and transport capability, and<br>5. pre-arrangement for analysis and interpretation. | | | | | | | |
| Post-Accident Analysis Capability (ONSITE) | 1) Gamme-ray Spectrum<br>2) pH<br>3) hydrogen<br>4) oxygen<br>5) boron | | | | | | | | | |

## M.) METEOROLOGY SYSTEM

The meteorology system contains the following measured variables:

(1) Wind Direction

(2) Wind Speed

(3) Vertical Temperature Difference

(4) Precipitation

Each of the four variables listed above are important in determining the transport of any radioactive effluent released from a post-accident nuclear power plant. The information gathered will be used for emergency planning, dose assessment to the public, and radioactive source estimates (both type and quantity released to the environment). Therefore, all four should be monitored on site. The entire meteorology system is listed in the following tables. Each table contains information taken from several reports and/or standards which deal with post-accident monitoring. This information is given in the tables on pages 8-9 of this report.

Additional comments and criticisms on each of the measured variables are given below:

<u>(1) Wind Direction</u>

a) This variable is definitly necessary for post-accident monitoring since it supplies information to civil defense, public officials, etc. about the path of any radioactive effluents released from the post-accident power plant.

b) The range is obvious! It must be 0° to 360°. The only question would be the accuracy (given as ± 5°) of this type equipment. A check with several manufacturers of modern weather forcasting equipment would clear up any problem in this area.

c) The type is fine.

(2) Wind Speed

a) This variable, in light of past Three-Mile Island, is definitely necessary to supply information about atmospheric conditions present near a post-accident nuclear power plant.

b) The range should be plant specific! A minimum speed could be specified but the maximum speed should be dependent upon local weather conditions. The accuracy seems acceptable but again several checks with manufactures of modern weather equipment could be made to clear up any doubts.

c) The type is fine (standard off-the-shelf items can be used).

(3) Vertical Temperature Difference

a) The vertical temperature difference gives emergency officials information about the atmospheric mixing present at the post-accident nuclear power plant. Hence, this variable is necessary in a post-accident environment.

b) The range seems arbitrary and needs further investigation.

c) The type is fine.

(4) Precipitation

a) Rain, snow, etc. can remove radioactive contaminants from the atmosphere. Hence, this variable is vital in a post-accident environment.

b) The range should be plant specific. Since each area of the country has different precipitation levels, it is necessary to check the local weather history to determine the maximum (single event) rainfall.

c) The type is fine.

SYSTEM: METEOROLOGY

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 4-12.74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| WIND DIRECTION | 0 TO 360° (±5° ACCURACY WITH A DEFLECTION OF 15°. STARTING SPEED 0.45 mps (1mph) | E | FOR DETERMINING AFFLUENT TRANSPORT DIRECTION FOR EMERGENCY PLANNING, DOSE ASSESSMENT, AND SOURCE ESTIMATES. | | | | | ✓ | | ✓ |
| WIND SPEED | 0 TO 30 mps (67 mph) (0.22 mps (0.5 mph) ACCURACY FOR WIND SPEED LESS THAN 11 mps (25 mph), WITH A STARTING THRESHOLD OF LESS THAN 0.45 mps (1 mph) | E | FOR DETERMINING EFFLUENT TRAVEL SPEED AND DILUTION FOR EMERGENCY PLANNING, DOSES, ASSESSMENTS, AND SOURCE ESTIMATES | | | | | ✓ | | ✓ |

# SYSTEM: METEOROLOGY (CONT.)

| MEASURED VARIABLE | RANGE | TYPE | PURPOSE | SAR REMARKS | BATTELLE BNI-X-647 | EPRI | NRC DRAFT RG 1.XX 6-12-74 | NRC ACRS 8-13-76 (CHIPMAN) | NRC LISTS TO ANS 4.5 7-30-79 | ANS 4.5 FIRST DRAFT |
|---|---|---|---|---|---|---|---|---|---|---|
| VERTICAL TEMPERATURE DIFFERENCE | −9°F TO +9°F (±0.3°F ACCURACY PER 164 FOOT INTERVAL | E | FOR DETERMINING EFFLUENT DIFFUSION RATES FOR EMERGENCY PLANNING, DOSE ASSESSMENTS AND SOURCE ESTIMATES | | | | | | | −5° TO +5°F |
| PRECIPITATION | RECORDING RAIN GAGE WITH A RANGE SUFFICIENT TO ASSURE ACCURACY OF TOTAL ACCUMULATION WITHIN 10% OF RECORDED VALUE – 0.01" RESOLUTION | E | FOR DETERMINING EFFLUENT TRANSPORT AND GROUND DEPOSITION FOR EMERGENCY PLANNING | | | | | | | |

SUMMARY OF THE RESULTS AND ANALYSIS

The following tables are a complete summary of the results of the analysis of each of the measured variables listed in the NRC report.

TABLE 2 PWR VARIABLES

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| CORE: | | | |
| Core Exit Temperature | 150°F to 2300°F | Yes | Needs additional studies. |
| Control Rod Position | Full in or not full in | Yes | Yes |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | Yes | Needs additional studies |
| REACTOR COOLANT SYSTEM: | | | |
| RCS Hot Leg Temperature | 150°F to 750°F | Yes | Yes |
| RCS Cold Leg Temperature | 150°F to 750°F | Yes | Yes |
| RCS Pressure | 15 psia to 4000psig | Yes | Needs additional studies |
| Pressurizer Level | Bottom tangent to top tangent | Yes | Yes |
| Degree of Subcooling | 200°F subcooling to 35°F superheat | Not absolutely necessary but useful | Needs additional studies |
| Reactor Core Level | Entire core plus safety margin above and below the core | Yes | Needs additional studies |
| Reactor Coolant Loop Flow | 0 to 120%)) design flow<br>-20% to 20%)) | Yes | Needs additional studies |
| Primary System Safety Relief Valve Positions or Flow Through or Pressure in Relief Valve lines | Closed-not closed | Yes | Yes |
| Radiation Level in Primary Coolant Water | 10 µCi/g to 10 Ci/g | Yes | Needs additional studies. |

TABLE 2 PWR VARIABLES

| Measured Variable | Range | Type | Is the type proper or not? |
|---|---|---|---|
| CORE: | | | |
| Core Exit Temperature | 150°F to 2300°F | B,C | Needs additional studies. |
| Control Rod Position | Full in or not full in | D | Yes |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | B | Yes |
| REACTOR COOLANT SYSTEM: | | | |
| RCS Hot Leg Temperature | 150°F to 750°F | B | Yes |
| RCS Cold Leg Temperature | 150°F to 750°F | B | Yes |
| RCS Pressure | 15 psia to 4000 psig | B,C | Yes |
| Pressurizer Level | Bottom tangent to top tangent | B,D | Yes |
| Degree of Subcooling | 200°F subcooling to 35°F superheat | B | Yes |
| Reactor Core Level | Entire core plus safety margin above and below the core | Needs to be assigned. | Needs additional studies. |
| Reactor Coolant Loop Flow | 0 to 120%) -20% to 20%) )design flow | B,D | Yes |
| Primary System Safety Relief Valve Position or Flow Through or Pressure in Relief Valve Lines | Closed-not closed | B,D | Yes |
| Radiation Level in Primary Coolant Water | 10 $\mu$Ci/g to 10Ci/g | C | Yes |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | Type | Is the type proper or not? |
| --- | --- | --- | --- |
| CONTAINMENT: | | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | B,C | Yes |
| Containment Atmosphere Temperature | 40°F to 400°F | B | Yes |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) | B,C | Yes |
| Containment Isolation Valve Position | Closed-not closed | B,C,D | Needs additional study. |
| Containment Sump Water Level | Narrow range(sump) Wide range (bottom of containment to 600,000 gallon level equivalent) | B,C | Yes |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr(60keV to 3 MeV photons with ±20% accuracy for photons of 0.1 to 3 MeV)($10^7$ R/hr for photons is approximately equivalent to $10^5$ rads per hour for betas and photons) | B,C | Yes |
| SECONDARY SYSTEMS: | | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | D | Yes |
| Steam Generator Level | From tube sheet to separators | D | Yes |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| CONTAINMENT: | | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | Yes | Needs additional studies. |
| Containment Atmosphere Temperature | 40°F to 400°F | Yes | Needs additional studies. |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) | Yes | Yes |
| Containment Isolation Valve Position | Closed-not closed | Yes | Yes |
| Containment Sump Water Level | Narrow Range (sump) Wide range (bottom of containment to 600,000 gallon level equivalent) | Yes | The range should be plant specific. |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr (60keV to 3 MeV photons with ± 20% accuracy for photons of 0.1 to 3 MeV) ($10^7$ R/hr for photons is approximately equivalent to $10^3$ rads per hour for betas and photons) | Yes | Needs additional studies. |
| SECONDARY SYSTEMS: | | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | Yes | Needs additional studies. |
| Steam Generator Level | From tube sheet to separators | Yes | The range should be plant specific. |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| SECONDARY SYSTEMS CONTINUED: | | | |
| Auxiliary Feedwater Flow | 0 to 110% design flow[1] | Yes | Yes |
| Main Feedwater Flow | 0 to 110% design flow[1] | Yes | Yes |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | No | |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^{5}$ $\mu Ci/cc$ | Yes | Needs additional studies |
| Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^{3}$ $\mu Ci/cc$ | Yes, but difficult to measure. | Needs additional studies. |
| AUXILIARY SYSTEMS: | | | |
| Containment Spray Flow | 0 to 110% design flow | Yes | No |
| Flow in HPI System | 0 to 110% design flow | Yes | Yes |
| Flow in LPI System | 0 to 110% design flow | Yes | Yes |
| Emergency Coolant Water Storage Tank Level | Top to bottom | Yes | Needs additional studies. |
| Pressure of $N_2$ in the accumulator Tank | To be determined later | Added | Added, needs additional studies. |
| Accumulator Tank Level | Top to bottom | Yes | Needs additional studies. |
| Accumulator Isolation Valve Positions | Closed-not closed | Yes | Yes |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | Type | Is the type proper or not? |
|---|---|---|---|
| SECONDARY SYSTEMS CONTINUED: | | | |
| Auxiliary Feedwater Flow | 0 to 110% design flow[1] | D | Yes |
| Main Feedwater Flow | 0 to 110% design flow[1] | E | Changed to D |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | E,D | Needs additional studies. |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc | None Listed | Assign Type C |
| Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc | E,C | Assign Type C |
| AUXILIARY SYSTEMS: | | | |
| Containment Spray Flow | 0 to 110% design flow | D | Yes |
| Flow in HPI System | 0 to 110% design flow | D | Yes |
| Flow in LPI System | 0 to 110% design flow | D | Yes |
| Emergency Coolant Water Storage Tank Level | Top to bottom | D | Yes |
| Pressure of $N_2$ in the Accumulator Tank | To be determined later | Assigned D | To be determined later; needs additional study. |
| Accumulator Tank Level | Top to bottom | D | Yes |
| Accumulator Isolation Valve Positions | Closed-not closed | C | Yes |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | | |
| RHR System Flow | 0 to 110% design flow | Yes | Yes |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F | Yes<br>Some questions exist | Needs additional study |
| Component Cooling Water Temperature | 32°F to 200°F | Yes | Needs additional study |
| Component Cooling Water Flow | 0 to 110% design flow | Yes | Yes |
| Flow in UHS Loop | 0 to 110% design flow | Yes | Yes |
| Temperature in Ultimate Heat Sink Loop | 30°F to 150°F | Yes, define more precisely | Needs additional study |
| Ultimate Heat Sink Level | Plant specific | Yes, define more precisly | Needs additional study |
| Heat Removal by the Containment Fan Coolers | Plant specific | No | Needs additional study |
| Boric Acid charging Flow | 0 to 110% design Flow | Yes, trouble with the purpose of this variable | Yes |
| Letdown Flow | 0 to 110% design flow | Yes | Yes |
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | No see text | Yes see text |
| RADWASTE SYSTEMS: | | | |
| High Level Radioactive Liquid Tank Level | Top to bottom | Yes | Should be plant specific |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% of design pressure | Yes | Should be plant specific |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | Type | Is the type proper or not? |
|---|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | | |
| RHR System Flow | 0 to 110% design flow | D | Yes |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F | D | Yes |
| Component Cooling Water Temperature | 32°F to 200°F | D | Yes |
| Component Cooling Water Flow | 0 to 110% design Flow | D | Yes |
| Flow in UHS Loop | 0 to 110% design flow | D | Yes |
| Temperature in Ultimate Heat Sink Loop | 30°F to 150°F | D | Yes |
| Ultimate Heat Sink Level | Plant specific | D | Yes |
| Heat Removal by the Containment Fan Coolers | Plant specific | P | No |
| Boric Acid Charging Flow | 0 to 110% design flow | B | Changed to D |
| Letdown Flow | 0 to 110% design flow | D | Yes |
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | D | Yes |
| RADWASTE SYSTEMS: | | | |
| High Level Radioactive Liquid Tank Level | Top to bottom | E | Upgraded to Type C or D |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% of design pressure | E | Upgraded to Type C or D |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| VENTILATION SYSTEMS: | | | |
| Emergency Ventilation Damper Positions | Open-closed status | Yes | Yes |
| Temperature of Space in Vicinity of Equipment Required for Safety | 70°F to 130°F | Yes | Needs additional study |
| POWER SUPPLIES: | | | |
| Status of Class 1E Power Supplies and Systems | Voltages and currents | Already done to some extent needs additional study | Needs additional study |
| Status of Non-class 1E Power Supplies and Systems | Voltages and currents | No | Needs additional study |
| RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT: | | | |
| Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors) | Yes | Needs additional study. |
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT: | | | |
| Effluent Radioactivity -Noble Gases | (Normal plus accident range for noble gas) | Yes, but needs additional study. Also needs further clarification. | Needs additional study |
| -Containment | $10^{-7}$ to $10^{5}$ $\mu$Ci/cc Xe-133 calibration | | |
| -Secondary Containment | $10^{-7}$ to $10^{4}$ $\mu$Ci/cc Xe-133 calibration | | |

TABLE 2 BWR VARIABLES continued

| Measured Variable | Range | Type | Is the type proper or not? |
|---|---|---|---|
| VENTILATION SYSTEMS: | | | |
| Emergency Ventilation Damper Position | Open-closed status | D | Yes |
| Temperature of space in Vicinity of Equipment Required for Safety | 30°F to 130°F | D | Yes |
| POWER SUPPLIES: | | | |
| Status of Class 1E Power Supplies and Systems | Voltages and currents | D | Yes |
| Status of Non-class 1E Power Supplies and Systems | Voltages and currents | E | Yes |
| RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT: | | | |
| Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors | E | Yes |
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT: | | | |
| Effluent Radioactivity -Noble Gases | (Normal plus accident range for noble gas) | E | Yes |
| -Containment | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc Xe-133 calibration | | |
| -Secondary Containment | $10^{-7}$ to $10^{4}$ $\mu$Ci/cc Xe-133 calibration | | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| **AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED:** | | | |
| -Auxiliary Buildings including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ $\mu$Ci/cc | | |
| -Other Release Points (including fuel handling area if separate from auxiliary building) | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc (permanently installed monitors) | | |
| Effluent Radioactivity -High Range Radiohalogens and Particulates | | Yes, but needs additional study. Also needs further clarification. | Needs additional study. |
| -Untreated Effluents | $10^{-3}$ to $10^{2}$ $\mu$Ci/cc | | |
| -HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to 10 $\mu$Ci/cc | | |
| -HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 $\mu$Ci/cc (permanently installed monitors) | | |
| Environs Radioactivity -High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60 keV to 3 MeV) (permanently installed monitors) | Yes, but needs additional study and further clarification. | Needs additional study. |
| Environs Radioactivity -Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ $\mu$Ci/cc for both radiohalogens and particulates (permanently installed monitors) | Yes, but needs additional study and further clarification | Needs additional study. |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | Type | Is the type proper or not? |
|---|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED: | | | |
| -Auxiliary Building including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ μCi/cc | | |
| -Other Release Points (including fuel handling area if separate from auxiliary building) | $10^{-7}$ to $10^{2}$ μCi/cc (permanently installed monitors) | | |
| -Effluent Radioactivity -High Range Radiohalogens and Particulates | | E | Yes |
| -Untreated Effluents | $10^{-3}$ to $10^{2}$ μCi/cc | | |
| -HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to 10 μCi/cc | | |
| -HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 μCi/cc (permanently installed monitors) | | |
| Environs Radioactivity -High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60 keV to 3MeV) (permanently installed monitors) | E | Yes |
| Environs Radioactivity Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ μCi/cc for both radiohalogens and particulates (permanently installed monitors) | E | Yes |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED: | | | |
| Plant and Environs Radioactivity (portable instruments) | Normal Range<br>0.1 to $10^4$ mR/hr photons<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc particulates<br>$10^{-9}$ to $10^{-4}$ iodine<br>High Range<br>0.1 to $10^4$ R/hr photons<br>0.1 to $10^4$ rads/hr betas and low energy photons<br>100-channel gamma-ray spectrometer | Yes, but needs additional study and clarification. | Needs additional study |
| POST-ACCIDENT SAMPLING CAPABILITY: | | | |
| Primary Coolant Sumps Containment Air | As required based on Reg Guide 1.4 guidelines | Yes | Needs additional study |
| POST-ACCIDENT ANALYSIS CAPABILITY (ONSITE): | 1. gamma-ray spectrum<br>2. pH<br>3. hydrogen<br>4. oxygen<br>5. boron | Yes | Needs additional study |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Type | Is the type proper or not? |
|---|---|---|---|
| **AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED:** | | | |
| Plant and Environs Radioactivity (portable instruments) | **Normal Range**<br>0.1 to $10^4$ mR/hr photons<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc particulates<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc iodine<br>**High Range**<br>0.1 to $10^4$ R/hr photons<br>0.1 to $10^4$ rads/hr betas and low energy photons<br>100-channel gamma-ray spectrometer | E | Yes |
| **POST-ACCIDENT SAMPLING CAPABILITY:** | | | |
| Primary Coolant Sumps Containment Air | As required based on Reg Guide 1.4 guidelines | N/A | Yes |
| **POST-ACCIDENT ANALYSIS CAPABILITY(ONSITE):** | 1.gamma-ray spectrum<br>2.pH<br>3.hydrogen<br>4.oxygen<br>5.boron | N/A | Yes |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Should the variable be measured or not? | Is the range proper or not? |
|---|---|---|---|
| METEOROLOGY: | | | |
| Wind Direction | 0 to 360° (±5° accuracy with a deflection of 15°. Startin speed 0.45mps(1 mph) | Yes | Yes |
| Wind Speed | 0 to 30 mps (67mph) (±0.22mps (0.5mph) accuracy for wind speed less than 11 mps (25mph), with a starting threshold of less than 0.45mps (1 mph) | Yes | Should be plant specific |
| Vertical Temperature Difference | -9°F to +9°F(±0.3°F accuracy per 164 foot intervals | Yes | Needs additional study. |
| Precipitation | Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value-0.01" resolution | Yes | Should be plant specific |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Type | Is the type proper or not? |
|---|---|---|---|
| **METEOROLOGY** | | | |
| Wind Direction | 0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1mph) | E | Yes |
| Wind Speed | 0 to 30 mps (67mph) (±0.22 mps (0.5 mph) accuracy for wind speed less than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1 mph) | E | Yes |
| Vertical Temperature Difference | -9°F to +9°F (±0.3°F accuracy per 164 foot intervals | E | Yes |
| Precipitation | Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value-0.01" resolution. | E | Yes |

CONCLUSIONS

Principal Author and Study Group Conclusions

(1) A complete summary of all of the conclusions in the analysis of nuclear regulatory guide 1.97 is given in The Summary of The Results and Analysis section of this report.

(2) The following general conclusions about the variables analyzed by the study group:

a) The study group agreed with most of the variables selected by the NRC for post-accidetn monitoring.

b) The ranges selected by the NRC lack documentation to justify the limits set forth in NucReg 1.97.

c) The NRC should be contacted to obtain information about the documentation used to justify the range settings. This information will be used by the study group to begin part two of this study.

d) The study group agreed with most of the qualification types assigned by the NRC.

e) Part two of this report will start with the range-set-point documentation supplied by the NRC. The study group will then proceed to analyze the NRC documentation, checking all major assumptions and models used by the NRC to reach the assigned set-points. Then a separate analysis will be conducted by the study group on the range set-points given in NucReg 1.97. Finally a comparison between the two values will be made and a report written summarizing the results.

(3) Nuclear Regulatory Guide 1.97 will increase the quality

and quantity of instrumentation used by the reactor operator to monitor an accident. This implies that the basic philosophy of the Nuclear Regulatory Commission is: The reactor operator is considered to be the principal means of protection during a post-accident environment. But this implies that the operator must be well informed! Thus the need for additional instrumentation set forth in NucReg. Guide 1.97.

The reactor industry, who design the automatic protective systems for nuclear power plants, rely more on automatic safety systems for the protection of the plants and less on the reactor operator. But this implies that the operator would be discouraged or even forbidden to act during an accident except to insure that the safety system did indeed come on in an emergency.

The utilities, because of the loss of large amounts of income when a reactor shuts down, tend to distrust automatic safety systems because the systems can frequently interfere with the utilities' desire to keep the plant on-line.

The public needs to be protected! Thus a basic conflict exists between the four different philosophies which are present today in the nuclear industry. The study group believes that the <u>Role of the Reactor Operator</u> should be precisely determined by a commission which consists of representatives from the NRC, industry, academe, etc. The commission could integrate the operator into a total safety system concept. Instead of being a questionable and undefined part of the reactor safety systems, the operator would become another engineered safety system o the plant. But this implies that the operator, inorder to meet the new guide-lines set up by the commision, would have to meet new education and training standards. Thus the simple act of defining the operator's function in the safety system, will produce the following results:

a) All three of the ma combatants in the operation of a nuclear power plant ((i) the NRC, (2) the nuclear industry, and (3) the utilities) will be working from only one philosophy in designing and upgrading safety systems.

b) The reactor operator's function will be completely determined.

c) The reactor safety systems (including the operator) will be designed with system concepts in mind.

d) The education and training of the reactor operator, inorder to meet the new definition of an operator, will be upgraded and maintained. This will insure that the safety system, as a whole, will be maintained at top efficiency (call this operator maintenance!).

e) With the operator becoming a part of the total safety system, designers of new plants must consider when desisning the control panel, etc. Thus human engineerin would be incorporated into the design of reactor control panel.

(4) Increasing the amount of information that the reactor operator must be aware of during an accident is not going to help the operator determine the nature of that accident. Even the world's best operator can become confused at times! Thus the only solution to this problem is data processoring by computers. The computer could analyze, simplify, etc. all the information about the nuclear power plant and display this information in an orderly manner. The <u>engineer</u> could determine, using his superior knowledge of the plant he designed, what parameters and their order of presentation would be seen by the reactor operator during an accident. Thus the reactor operator would be forced to deal with the most important safety problems in a logical manner.

An example would be the accident at Three-Mile Island. The computer would print out (the use of TV screens is recommended by the study group) data in a format given below:

Temp. on the other side of the safety relief valve on the pressurizer = 235° <u>HIGH</u> possible causes:

a) small loca in progress

b) etc.

Main feedwater system shutdown

Block valve shut on the auxiliary feedwater system

<u>NO FLOW INTO STEAM GENERATOR</u>

Saturation temp. and pressure readout on a continuing basis

Etc.

Beyond simple data reduction and simplification the computer could "add " up all the data and actually give the cause of the problem and take corrective action. But this is in the future and simple data reduction would solve many problems of reactor control.

INDIVIDUAL STUDY GROUP - E.REP CONCLUSIONS

NRC has done a commendable task by enlisting 70 variables required for post-accident monitoring. However, with the momentous task of reading, interpreting, and correlating the information content of these variables confronting an operator it is pertinent to question the role that an operator is supposed to execute, how much freedom he has in executing control actions and what is the degree of efficiency (quality of performance) expected from him. Since it is humanly not possible to comprehend the information content of 70 variables is it necessary to display all of them? If not, then what is the minimum information he should be provided? These questions must be answered before any conclusions about proper post-accident instrumentation can be made!

The addition of all the variables called for by Reg. Guide 1.97 Rev. 2 would increase greatly the number of instruments that the operator will have to be aware of if the instrumentation is to be located in the control room. The operator will become overloaded or saturated so that some form of computer data reduction and presentation would be a great help in ordering and prioritizing data.

APPENDIX A

# TABLE 2 PWR VARIABLES

| Measured Variable | Range | Comments |
|---|---|---|
| CORE: | | |
| Core Exit Temperature | 150°F to 2700°F | Variable needs to be measured but could change Type letter to E or C; place two thermocouples in each well for range; range next quarter |
| Control Rod Position | Full in or not full in | Variable needs to be measured; range is fine |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | Variable needs to be measured; range next quarter |
| REACTOR COOLANT SYSTEM: | | |
| RCS Hot Leg Temperature | 150°F to 750°F | Variable needs to be measured; range next quarter |
| RCS Cold Leg Temperature | 150°F to 750°F | Variable needs to be measured; range next quarter |
| RCS Pressure | 15 psia to 4000psig | Variable needs to be measured; range next quarter |
| Pressurizer Level | Bottom tangent to top tangent | Variable needs to be measured; range is fine |
| Degree of Subcooling | 200°F subcooling to 75°F superheat | Very difficult to do; can't be done directly; what about localized boiling range next quarter |
| Reactor Core Level | Entire Core plus safety levels above and below | Add this variable; difficult to do; look in how it is done on a PWR |
| Reactor Coolant Loop Flow | 0 to 120%<br>-20% to 20% )design flow 1 | What type of flow units would be used? Do they measure flow in both directions. Actual flow or pump speeds? Range Some problems here |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | Comments |
|---|---|---|
| REACTOR COOLANT SYSTEM CONTINUED: | | |
| Primary System Safety Relief Valve Positions or Flow Through or Pressure in Relief Valve Lines | Closed-not closed | Do you measure the actual valve position or the electrical or hydralic system conditions? Needs discussion |
| Radiation Level in Primary Coolant Water | 10 $\mu$Ci/g to 10 Ci/g | Variable should be measured range next quarter. Should sampling be done continuously or by using sampling |
| CONTAINMENT: | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | Variable should be measured; need more information on (a) locations and (b) how many and how many at each location. Range next quarter |
| Containment Atmosphere Temperature | 40°F to 400°F | Variable should be measured Variable Type definite E Range next quarter |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) | Variable should be measured; Range should be capable of operating up to containment rupture. |
| Containment Isolation Valve Position | Closed-not closed | Variable should be measured; Range is fine |
| Containment Sump Water Level | Narrow range (sump) Wide range (bottom of containment to 600,000 gallon level equivalent) | Variable should be measured; range should be plant specific |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
|---|---|---|
| CONTAINMENT CONTINUED: | | |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr (60 keV to 3 MeV photons with ±20% accuracy for photons of 0.1 to 3 MeV) ($10^7$ R/hr for photons is approximately equivalent to $10^3$ rads per hour for betas and photons) | Variable should be measured; Range next quarter |
| SECONDARY SYSTEMS: | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | Variable should be measured; make sure on the range that the normal secondary pressure goes at least 20% beyond |
| Steam Generator Level | From tube sheet to separators | Variable should be measured; range is fine |
| Auxiliary Feedwater Flow | 0 to 110% design flow[1] | Variable should be measured; range is fine |
| Main Feedwater Flow | 0 to 110% design flow[1] | Variable should be measured; range is fine |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | Not needed; if they open the noise will let operator know |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^2$ $\mu$Ci/cc | Variable should be measured range too high; 2 or 3 orders of magnitude are good |
| Radioactivity in Effluent from Steam Generated Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^2$ $\mu$Ci/cc | Difficult to do; could use two monitor in the main steam lines instead of outside the secondary; range [illegible] next quarter (too high) |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
|---|---|---|
| AUXILIARY SYSTEMS: | | |
| Containment Spray Flow | 0 to 110% design flow | Variable should be measured. How do you indicate this on-off flow or percent meter flow? |
| Flow in HPI System | 0 to 110% design flow | Variable should be measured; range is fine |
| Flow in LPI System | 0 to 110% design flow | Variable should be measured; range is fine |
| Emergency Coolant Water Storage Tank Level | Top to bottom | Variable should be measured; range good except it should go from top to operating set-level |
| Pressure in $N_2$ | See S.A.P.S. for details | Already done |
| Accumulator Isolation Valve Positions | Closed-not closed | Variable should be measured; range is fine |
| Accumulator Tank Level | Top to bottom | Already measured; present instrumentation is fine |
| RHR System Flow | 0 to 110% design flow | Variable should be measured;) Already<br>Range is fine ) done |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F<br>Range next quarter | Variable should be measured; which out temperature (1) out of reactor or (2) out of coolant heat exchanger |
| Component Cooling Water Temperature | 32°F to 200°F<br>Range next quarter | |
| Component Cooling Water Flow | 0 to 110 design flow | Variable should be measured; range is fine |
| Flow in UHS Loop | 0 to 110% design flow | Variable should be measured; range is fine |
| Temperature in Ultimate Heat Sink Loop | 30°F to 150°F | Which Temperature do you mean: inlet or outlet??<br>Range next quarter |
| Ultimate Heat Sink Level | Plant specific | Variable should be measured; range is fine |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
|---|---|---|
| AUXILIARY SYSTEMS continued: | | |
| Heat Removal by the Containment Fan Coolers | Plant specific | Problem area; Kiper had major point (see him) Range is fine. |
| Boric Acid Charging Flow | 0 to 110% design flow | Question with the purpose of this variable. Question with the type. Range is fine. |
| Letdown Flow | 0 to 110% design flow[1] | Variable should be measured; question with the purpose (RCP's); should put partial information; Type D ok; range is fine. |
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | Questionable; needs to be more specific or plant specific; needs work |
| RADWASTE SYSTEMS: | | |
| High Level Radioactive Liquid Tank Level | Top to Bottom | What tank? Should specify what tank! Question with the purpose The entire primary water inventory? Type should be changed to type E |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% of design pressure[2] | Type should be changed to type E |
| VENTILATION SYSTEMS: | | |
| Emergency Ventilation Damper Position | Open-closed status | Already exists |
| Temperature of Space in Vicinity of Equipment Required for Safety | 30°F to 150°F Range next quarter | Type should be changed to type E; Specific to plant the number should be made as small as possible containment temperature more important; Auxiliary building not as critical |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
|---|---|---|
| POWER SUPPLIES: | | |
| Status of Class 1E Power Supplies and Systems | Voltages and currents | Variable should be measured; should on-off or quantitive meters be used; very large numbers required; does cost justify information |
| Status of Non-Class 1E Power Supplies and Systems | Voltages and currents | Questionable; if you can't justify Class 1E, what about the Non-class 1E. |
| RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT | | |
| Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors) | Variable should be measured; the high range is questionable; is this in addition to the normal range? Range next quarter |
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT: | | |
| Effluent Radioactivity -Noble Gases | (Normal plus accident range for noble gas) | Not specific enough? What do they mean? Many detectors in many places or single detector in the optimum places? Many questions? Range next quarter |
| Containment | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc Xe-133 calibration | |
| Secondary Containment | $10^{-7}$ to $10^{4}$ $\mu$Ci/cc Xe-133 calibration | |
| Auxiliary Building including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ $\mu$Ci/cc | |
| Other release Points including fuel handling area if separate from auxiliary building) | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc (permanently installed monitors) | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
| --- | --- | --- |
| **AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED:** | | |
| Effluent Radioactivity High Range Radiohalogens and Particulates | | Same as above<br>Range next quarter |
| Untreated Effluents | $10^{-3}$ to $10^{2}$ $\mu$Ci/cc | |
| HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to 10 $\mu$Ci/cc | |
| HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 $\mu$Ci/cc (permanently installed monitors) | |
| Environs Radioactivity-- High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60 keV to 3 MeV) (Permanently installed monitors)<br>Ranges Next quarter | Very expensive; is this a local readout (at the monitor location) or is this a remote readout (in the reactor operating room); do you mean wires going back to plant civil defense? |
| Environs Radioactivity - Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ $\mu$Ci/cc for both radiohalogens and particulates (permanently installed monitors) | Very expensive; same as above; ranges next quarter |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASES FROM THE PLANT CONTINUED: | | |
| Plant and Environs Radioactivity (portable instruments) | Normal Range<br>0.1 to $10^4$ mR/hr photons<br>$10^{-9}$ to $10^{-4}$ μCi/cc particulates<br>$10^{-9}$ to $10^{-4}$ μCi/cc iodine<br>High Range<br>0.1 to $10^4$ R/hr photons<br>0.1 to $10^4$ rads/hr betas and low energy photons<br>100-channel gamma-ray spectrometer | Variable should be measured; ranges next quarter |
| POST-ACCIDENT SAMPLING CAPABILITY: | | |
| Primary Coolant<br>Sumps<br>Containment Air | As required based on Reg guide 1.4 guidelines | Check out Reg guide 1.4 |
| POST ACCIDENT ANALYSIS CAPABILITY (ONSITE): | 1. gamma-ray spectrum<br>2. pH<br>3. hydrogen<br>4. oxygen<br>5. boron | Variable should be measured; ranges are plant specific |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Comments |
| --- | --- | --- |
| METEOROLOGY: | | |
| Wind Direction | 0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1mph ) | Variable should be measured; range is fine |
| Wind Speed | 0 to 30 mps (67 mph) (±0.22 mps (0.5 mph) accuracy for wind speed less than 11 mps (25 mph), with a starting threshold of less than 0.45mps (1 mph) | Variable should be measured; range should be site specific. |
| Vertical Temperature Difference | -9°F to +9°F (±0.3°F accuracy per 164 foot intervals | Variable should be measured; why 164 feet |
| Precipitation | Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value - 0.01" resolution | Variable should be measured; range should be site specific. |

TABLE 2 PWR VARIABLES

| Measured Variable | Range | Battelle | BMI-X-647 |
|---|---|---|---|
| CORE: | | | |
| Core Exit Temperature | 150°F to 2300°F | | |
| Control Rod Position | Full in or not full in | X | |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | X | |
| REACTOR COOLANT SYSTEM: | | | |
| RCS Hot Leg Temperature | 150°F to 750°F | X | Listed under one heading ("Primary Coolant Temperature (PWR)) in the Battelle report |
| RCS Cold Leg Temperature | 150°F to 750°F | X | |
| RCS Pressure | 15 psia to 4000 psig | X | |
| Pressurizer Level | Bottom tangent to top tangent | X | When the liquid level falls below the pressurizer elevation its level is unknown!! Look into direct reactor core level measurement!? |
| Degree of Subcooling | 200°F subcooling to 35°F superheat | | |
| Reactor Coolant Loop Flow | 0 to 120% ) design<br>- 20% to 20% ) flow | | |
| Primary System Safety Relief Valve Positions or Flow Through or Pressure in Relief Valve Lines | Closed-not closed | | |
| Radiation Level in Primary Coolant Water | 10 μCi/g to 10 Ci/g | | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle PNL-Y-647 | |
|---|---|---|---|
| CONTAINMENT: | | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | Y | |
| Containment Atmosphere Temperature | 40°F to 400°F | Y | |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) | Y | |
| Containment Isolation Valve Position | Closed-not closed | | |
| Containment Sump Water Level | Narrow range (sump) Wide range (bottom of containment to 600,000 gallon level equivalent | Y | |
| High Range Containment Area Radiation | 1 to $10^{7}$ R/hr (60 keV to 3 MeV photons with $\pm$20% accuracy for photons of 0.1 to 3 MeV)($10^{7}$ R/hr for photons is approximately equivalent to $10^{3}$ rads per hour for betas and photons | Y | What about low range? Battelle report only deals with "increased radiation levels". |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle BMI-X-647 |
|---|---|---|
| SECONDARY SYSTEMS: | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | X |
| Steam Generator Level | From tube sheet to separators | X |
| Auxiliary Feedwater Flow | 0 to 110% design Flow[1] | X |
| Main Feedwater Flow | 0 to 110% design flow[1] | X |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^{5}$ $\mu$Ci/cc | X |
| Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^{5}$ $\mu$Ci/cc | X |
| AUXILIARY SYSTEMS: | | |
| Containment Spray Flow | 0 to 110% design flow | |
| Flow in HPI System | 0 to 110% design flow | |
| Flow in LPI System | 0 to 110% design flow | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle PNI-X-647 |
|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | |
| Emergency Coolant Water Storage Tank Level | Top to bottom | X |
| Accumulator Tank Level | Top to bottom | |
| Accumulator Isolation Valve Positions | Closed-not closed | |
| RHR System Flow | 0 to 110% design flow | |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F | |
| Component Cooling Water Temperature | 32°F to 200°F | |
| Component Cooling Water Flow | 0 to 110% design flow | |
| Flow in UHS Loop | 0 to 110% design flow | |
| Temperature in Ultimate Heat Sink Loop | 30°F to 150°F | |
| Ultimate Heat Sink Level | Plant specific | |
| Heat Removal by the Containment Fan Coolers | Plant specific | |
| Boric Acid Charging Flow | 0 to 110% design flow | |
| Letdown Flow | 0 to 110% design flow | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle BNI-X-647 |
|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | |
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | |
| RADWASTE SYSTEMS: | | |
| High Level Radioactive Liquid Tank Level | Top to bottom | |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% of design pressure[2] | |
| VENTILATION SYSTEMS: | | |
| Emergency Ventilation Damper Position | Open-closed status | |
| Temperature of Space in Vicinity of Equipment Required for Safety | 50°F to 150°F | X |
| POWER SUPPLIES: | | |
| Status of Class 1E Power Supplies and Systems | Voltages and currents | X Battelle report just lists: "Status of power supplies" most likely concerned about safety related power supplies |
| Status of Non-Class 1E Power Supplies and Systems | Voltages and currents | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle PNI-X-647 |
|---|---|---|
| <u>RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT:</u> | | |
| Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors) | |
| <u>AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT:</u> | | |
| Effluent Radioactivity Noble Gases | (Normal plus accident range for noble gas) | |
| Containment | $10^{-7}$ to $10^{5}$ $\mu Ci/cc$ Xe-133 calibration | |
| Secondary Containment | $10^{-7}$ to $10^{4}$ $\mu Ci/cc$ Xe-133 calibration | |
| Auxiliary Building including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ $\mu Ci/cc$ | |
| Other Release points (including fuel handling area if separate from auxiliary building | $10^{-7}$ to $10^{2}$ $\mu Ci/cc$ (permanently installed monitors) | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle [illegible]-647 |
|---|---|---|
| **AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED:** | | |
| Effluent Radioactivity - High Range Radiohalogens and Particulates | | |
| Untreated Effluents | $10^{-3}$ to $10^{2}$ $\mu$Ci/cc | |
| HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to 10 $\mu$Ci/cc | |
| HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 $\mu$Ci/cc (permanently installed monitors) | |
| Environs Radioactivity - High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60 keV to 3 MeV) (permanently installed monitors) | |
| Environs Radioactivity - Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ $\mu$Ci/cc for both radiohalogens and particulates (permanently installed monitors) | |
| Plant and Environs Radioactivity (portable instruments) | Normal Range<br>0.1 to $10^{4}$ mR/hr photons<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc particulates<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc iodine | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle BMI-X-647 |
|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED: | | |
| Plant and Environs Radioactivity (portable instruments) | High Range<br>0.1 to $10^4$ R/hr photons<br>0.1 to $10^4$ rads/hr betas and low energy photons<br>100-channel gamma-ray spectrometer | |
| POST-ACCIDENT SAMPLING CAPABILITY: | | |
| Primary Coolant<br>Sumps<br>Containment Air | As required based on Reg guide 1.4 guidelines | |
| POST-ACCIDENT ANALYSIS CAPABILITY (ONSITE): | 1. gamma-ray spectrum<br>2. pH<br>3. hydrogen<br>4. oxygen<br>5. boron | |
| METEOROLOGY: | | |
| Wind Direction | 0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45mps (1 mph) | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Battelle BMI-X-647 |
|---|---|---|
| METEOROLOGY CONTINUED: | | |
| Wind Speed | 0 to 30 mps (67 mph) (±0.22 mps(0.5 mph) accuracy for wind speed less than 11 mps (25 mph), with a starting threshold of less than 0.45 mph (1 mph) | |
| Vertical Temperature Difference | -9°F to +9°F (±0.3°F accuracy per 164 foot intervals | |
| Precipitation | Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value - 0.01" resolution | |
| MISCELLANEOUS: | | |
| Pump Speeds | | X Very vague! What pumps should be monitored? etc. |
| Boron Concentration | | X Measures shutdown or subcriticality of the reactor in conjunction with the control rod position. |
| Area Radiation Levels in Auxiliary Buildings | | X Must be made more specific |
| Off-Site Radiation | | X Must be made more specific |
| Valve Position | | X What valves? Must be made more specific. |

# TABLE 2 PWR VARIABLES

| Measured Variable | Range | Type | Remarks |
|---|---|---|---|
| CORE: | | | |
| Core Exit Temperature | 150°F to 2300°F | B,C | No mention in PSAR, FSARS<br>Very large range<br>2 thermocouples required to cover range<br>(0-530°F)(530-2300°F) |
| Control Rod Position | Full in or not full in | D | Not mentioned in PSAR & FSARS but at least for B&W plants in limit and 0% indication given in control room |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | E | Range adequately covered in DP, FSAR not mentioned in PSARS |
| REACTOR COOLANT SYSTEM: | | | |
| RCS Hot Leg Temperature | 150°F to 750°F | B | Sensor can cover range, but will need more readout meters<br>Indicator Ranges<br>(520-620°F D-B ±2% FR)<br>(0-700°F Watts Bar ±4% FR)<br>(50-650°F B&W STDPLT ±5% FR)<br>(475-625°F Comb Engr Std No Acc) |
| RCS Cold Leg Temperature | 150°F to 750°F | B | Same as above. Industry available Sigma 375-675°F ±1% |
| RCS Pressure | 15 psia to 4000 psig | B,C | Upper limit range seems arbitrary should be approx. equal to bursting pressure of RCS. Will require two separate instruments to cover range. Bourdon Tube can cover upper range 0-3000psig Watts Bar 0-2500psig D B FS R |
| Pressurizer Level | Bottom tangent to top tangent | B,D | Can be done |
| Degree of Subcooling | 200°F subcooling to 35°F superheat | B | Can be done |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Type | Remarks |
|---|---|---|---|
| REACTOR COOLANT SYSTEM CONTINUED: | | | |
| Reactor Coolant Loop Flow | 0 to 120%) ) design flow[1]<br>-20% to 20%) | B,D | 0-120% most probably can be done<br>-20%-20% not mentioned in SAPS |
| Primary System Safety Relief Valve Positions or Flow Through or Pressure in Relief Valves | Closed-not closed | B,D | Required by lessons learned Short Term Recommendations (2.1.3.a) |
| Radiation Level in Primary Coolant Water | 10 µCi/g to 10 Ci/g | C | (Later) |
| CONTAINMENT: | | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete: 4 times design pressure for steel | B,C | 0-60 psia DB FSAR with 36psia design; [illegible] 4-55 psia |
| Containment Atmosphere Temperature | 40°F to 400°F | B | (Later) |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) | B,C | D B FSAR 0-5% ±2% FS |
| Containment Isolation valve Position | Closed-not closed | B,C,D | Can be done per D B FSAR |
| Containment Sump Water Level | Narrow range (sump)<br>Wide Range (bottom of containment to 600,000 gallon level equivalent) | B,C | Sump yes;<br>Wide range is possible (see TMI-2 sequence of events) |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Type | Remarks |
|---|---|---|---|
| **CONTAINMENT CONTINUED:** | | | |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr (60 keV to 3 MeV photons with ±20% accuracy for photons of 0.1 to 3 MeV)($10^7$ R/hr for photons is approximately equivalent to $10^5$ rads per hour for betas and photons) | B,C | None listed in SARS<br>General Atomic 1- $10^8$ R/hr |
| **SECONDARY SYSTEMS:** | | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | | N/A |
| Steam Generator Level | From tube sheet to separators | | N/A |
| Auxiliary Feedwater Flow | 0 to 110% design flow[1] | | N/A |
| Main Feedwater Flow | 0 to 110% design flow[1] | | N/A |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | | N/A |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^5$ $\mu$Ci/cc | | N/A |
| Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^5$ $\mu$Ci/cc | | N/A |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Type | Remarks |
|---|---|---|---|
| RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT: | | | |
| Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors) | E | Kaman Science Mod [illegible] $10^{-1}$ - $10^{5}$ R/hr |
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT: | | | |
| Effluent Radioactivity - Noble Gases | (Normal plus accident range for noble gas) | E | Kaman Science Mod [illegible] [illegible] $10^{-7}$ to $10^{[illegible]}$ [illegible]Ci/cc |
| -Containment | $10^{-7}$ to $10^{2}$ μCi/cc Xe-133 calibration | | |
| -Secondary Containment | $10^{-7}$ to $10^{4}$ μCi/cc Xe-133 calibration | | |
| -Auxiliary Building including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ μCi/cc | | |
| -Other Release points (including fuel handling area if separate from auxiliary building | $10^{-7}$ to $10^{2}$ μCi/cc (Permanently installed monitors) | | |
| Effluent Radioactivity - High Range Radiohalogens and Particulates | | | |
| Untreated Effluents | $10^{-3}$ to $10^{2}$ μCi/cc | E | Kaman Science KDA-1 (photons) $10^{-3}$ - $10^{2}$ R/hr |
| Re[illegible] Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to 10 μCi/cc | | |

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TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | Type | Remarks |
|---|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED: | | | |
| HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 μCi/cc (permanently installed monitors) | | |
| Environs Radioactivity High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60 keV to 3 MeV) (permanently install- monitors) | E | Victoreen 845 $10^{-4}$ to $10^{4}$ R/hr |
| Environs Radioactivity Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ μCi/cc for both radiohalogens and particulates (permanently installed monitors) | E | Kaman KPIM $10^{-9}$ - $10^{-3}$ Ci/cc |
| Plant and Environs Radioactivity (portable instruments) | Normal Range<br>0.1 to $10^{4}$ mR/hr photons | E | Kaman KDA-1 $10^{-1}$ - $10^{4}$ R/hr |
| | $10^{-9}$ to $10^{-4}$ μCi/cc particulates | E | Kaman DPM $10^{-9}$ - $10^{-4}$ Ci/cc |
| | $10^{-9}$ to $10^{-4}$ μCi/cc iodine | E | Kaman KIM $10^{-9}$ - $10^{-4}$ Ci/cc |
| | High Range<br>0.1 to $10^{4}$ R/hr photons | E | Kaman KHPA-4 .1- $10^{4}$ R/hr |
| | 0.1 to $10^{4}$ rads/hr betas and low energy photons | E | Kaman KHPA-4 .1- $10^{4}$ R/hr |
| | 100-channel gamma-ray spectrometer | | ? |

TABLE 2 PWR VARIABLES

| Measured Variable | Range | EPRI |
|---|---|---|
| CORE: | | |
| Core Exit Temperature | 150°F to 2300°F | |
| Control Rod Position | Full in or not full in | X |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | X |
| REACTOR COOLANT SYSTEM: | | |
| RCS Hot Leg Temperature | 150°F to 750°F | X |
| RCS Cold Leg Temperature | 150°F to 750°F | |
| RCS Pressure | 15 psig to 4000 psig | X |
| Pressurizer Level | Bottom tangent to top tangent | X |
| Degree of Subcooling | 200°F subcooling to 75°F superheat | |
| Reactor Vessel Water Level | | X |
| Reactor Coolant Loop Flow | 0 to 120%) ) design<br>-20% to 20%) flow | X |
| Primary System Safety Relief Valve Positions, or Flow Through or Pressure In Relief Valve Lines | Closed-not closed | |
| Radiation Level in Primary Coolant Water | 10 μCi/g to 10 Ci/g | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| CONTAINMENT: | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | X |
| Containment Atmosphere Temperature | 40°F to 400°F | |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2] | |
| Containment Isolation Valve Position | Closed-not closed | |
| Containment Sump Water Level | Narrow range(sump) Wide range (bottom of containment to 600,000 gallon level equivalent) | |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr(60keV to 3 MeV photons with ±20% accuracy for photons of 0.1 to 3 MeV)($10^7$ R/hr for photons is approximately equivalent to $10^3$ rads per hour for betas and photons) | |
| SECONDARY SYSTEMS: | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| SECONDARY SYSTEMS CONTINUED: | | |
| Steam Generator Level | From tube sheet to separators | |
| Auxiliary Feedwater Flow | 0 to 110% design flow[1] | X |
| Main Feedwater Flow | 0 to 110% design flow[1] | |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc | |
| Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc | |
| AUXILIARY SYSTEMS: | | |
| Containment Spray Flow | 0 to 110% design flow | X |
| Flow in HPI System | 0 to 110% design flow | X |
| Flow in LPI System | 0 to 110% design flow | X |
| Emergency Coolant Water Storage Tank Level | Top to bottom | |
| Accumulator Tank Level | Top to bottom | |
| Accumulator Isolation Valve Positions | Closed-not closed | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | |
| RHR System Flow | 0 to 110% design flow | X |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F | |
| Component Cooling Water Temperature | 32°F to 200°F | |
| Component Cooling Water Flow | 0 to 110% design flow | |
| Flow in UHS Loop | 0 to 110% design flow | |
| Temperature in Ultimate Heat Sink Loop | 30°F to 150°F | |
| Ultimate Heat Sink Level | Plant specific | |
| Heat Removal by the Containment Fan Coolers | Plant specific | |
| Boric Acid Charging Flow | 0 to 110% design flow | X |
| Letdown Flow | 0 to 110% design flow[1] | |
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | |
| RADWASTE SYSTEMS: | | |
| High Level Radioactive Liquid Tank Level | Top to bottom | |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% of design pressure[2] | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| <u>VENTILATION SYSTEMS:</u><br>Emergency Ventilation Damper Position | Open-closed status | X |
| Temperature of Space in Vicinity of Equipment Required for Safety | 70°F to 130°F | |
| <u>POWER SUPPLIES:</u><br>Status of Class 1E Power Supplies and Systems | Voltages and currents | |
| Status of Non-Class 1E Power Supplies and Systems | Voltages and currents | |
| <u>RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT:</u><br>Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors) | |
| <u>AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT:</u><br>Effluent Radioactivity - Noble Gases<br>-Containment | (normal plus accident range for noble gas)<br>$10^{-7}$ to $10^{5}$ $\mu$Ci/cc Xe-133 calibration | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| **AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED:** | | |
| -Secondary Containment | $10^{-7}$ to $10^{4}$ $\mu$Ci/cc Xe-133 calibration | |
| -Auxiliary Building including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ $\mu$Ci/cc | |
| -Other Release Points (including fuel handling area if separate from auxiliary building | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc (permanently installed monitors) | |
| Effluent Radioactivity -High Range Radiohalogens and Particulates | | |
| Untreated Effluents | $10^{-3}$ to $10^{2}$ $\mu$Ci/cc | |
| HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to 10 $\mu$Ci/cc | |
| HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 $\mu$Ci/cc (permanently installed monitors) | |
| Environs Radioactivity High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60keV to 3 MeV) (permanently installed monitors) | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASES FROM THE PLANT CONTINUED: | | |
| Environs Radioactivity -Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ $\mu$Ci/cc for both radiohalogens and Particulates (permanently installed monitors) | |
| Plant and Environs Radioactivity (portable instruments) | Normal Range<br>0.1 to $10^4$ mR/hr photons<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc particulates<br>$10^{-9}$ to $10^{-4}$ $\mu$Ci/cc iodine<br>High Range<br>0.1 to $10^4$ R/hr photons<br>0.1 to $10^4$ rads/hr betas and low energy photons<br>100-channel gamma-ray spectrometer | |
| POST-ACCIDENT SAMPLING CAPABILITY: | | |
| Primary Coolant Sumps Containment Air | As required based on Reg Guide 1.4 guidelines | |
| POST-ACCIDENT ANALYSIS CAPABILITY (ONSITE): | 1. gamma-ray spectrum<br>2. pH<br>3. hydrogen<br>4. oxygen<br>5. boron | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | EPRI |
|---|---|---|
| METEOROLOGY: | | |
| Wind Direction | 0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps(1mph) | |
| Wind Speed | 0 to 30 mps (67 mph) (±0.22 mps(0.5 mph) accuracy for wind speed less than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1mph) | |
| Vertical Temperature Difference | -9°F to +9°F (±0.3°F accuracy per 164 foot intervals | |
| Precipitation | Recording rain gage with range sufficient to assure accuracy of total accumulation of within 10% of recorded value--0.01" resolution | |

# TABLE 2 PWR VARIABLE

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chipman) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| **CORE:** | | | | | |
| Core Exit Temperature | 150°F to 2300°F | | | X | X (150-2200°F) |
| Control Rod Position | Full in or not full in | X | X | X | X |
| Neutron Flux | 1 c/s to 1% power (at least one fission counter) | X | X | X | shutdown to 10% full power |
| **REACTOR COOLANT SYSTEM:** | | | | | |
| RCS Hot Leg Temperature | 150°F to 750°F | X | X | X | 150 to 900 |
| RCS Cold Leg Temperature | 150°F to 750°F | X | X | X | 150 to 900 |
| RCS Pressure | 15 psia to 4000psig | X | X | X | 15 psig to [illegible] design pressure |
| Pressurizer Level | Bottom Tangent to top tangent | X | X | X | |
| Degree of Subcooling | 200°F subcooling to 35°F superheat | | | X | 200°F to 5°F superheat |
| Reactor Coolant Loop Flow | 0 to 120%) -20% to 20%) design flow | X | X | X | 0 to 120% 0 to 20% |
| Primary System Safety Relief Valve Position or Flow Through or Pressure In Relief Valve Lines | Closed-not closed | X | | X | X |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chipman) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| REACTOR COOLANT SYSTEM CONTINUED: | | | | | |
| Radiation Level in Primary Coolant Water | 10 $\mu$Ci/g to 10 Ci/g | X | | X | X |
| CONTAINMENT: | | | | | |
| Containment Pressure | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | X | X | X | X |
| Containment Atmosphere Temperature | 40°F to 400°F | X | X | X | X |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2] | X | X | X | X |
| Containment Isolation Valve Position | Closed-not closed | | | X | X |
| Containment Sump Water Level | Narrow range (sump) Wide range (bottom of containment to 600,000 gallon level equivalent) | X | X | X | |
| High Range Containment Area Radiation | 1 to $10^7$ R/hr(60keV to 3 MeV photons with ±20% accuracy for photons of 0.1 to 3 MeV)($10^7$ R/hr for photons is approximately equivalent to $10^8$ rads per hour for betas and photons) | X | X | X | |

TABLE 2 PWR VARIABLES Continued

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chipman) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| SECONDARY SYSTEMS: | | | | | |
| Steam Generator Pressure | From pressure for safety valve setting to plus 20% of safety valve setting | X | X | X | 0 to 110% |
| Steam Generator Level | From tube sheet to separators | X | X | X | X |
| Auxiliary Feedwater Flow | 0 to 110% design flow[1] | X | | X | X |
| Main Feedwater Flow | 0 to 110% design flow[1] | | | X | X |
| Safety/Relief Valve Positions or Main Steam Flow | Closed-not closed | | | X | |
| Radiation in Condenser Air Removal System | $10^{-7}$ to $10^{5}$ μCi/cc | X | X | | X |
| Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-7}$ to $10^{5}$ μCi/cc | | | | X |
| AUXILIARY SYSTEMS: | | | | | |
| Containment Spray Flow | 0 to 110% design flow | | X | X | X |
| Flow in HPI System | 0 to 110% design flow | | | | X |
| Flow in LPI System | 0 to 110% design Flow | | | | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chirman) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | | | | |
| Emergency Coolant Water Storage Tank Level | Top to bottom | X | X | X | X |
| Accumulator Tank Level | Top to bottom | | | X | X |
| Accumulator Isolation Valve Positions | Closed-not closed | | | | X |
| RHR System Flow | 0 to 110% design flow | | | X | X |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F | | | | 32°-150°F |
| Component Cooling Water Temperature | 32°F to 200°F | | | X | 32°-130°F |
| Component Cooling Water Flow | 0 to 110% design flow | | | X | X |
| Flow in UHS Loop | 0 to 110% design flow | | | X | X |
| Temperature in Ultimate Heat Sink Loop | 30°F to 150°F | | X | X | 30°-130° |
| Ultimate Heat Sink Level | Plant specific | | X | X | X |
| Heat Removal by the Containment Fan Coolers | Plant specific | | | | X |
| Boric Acid Charging Flow | 0 to 110% design flow | | | X | X |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chipman) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| AUXILIARY SYSTEMS CONTINUED: | | | | | |
| Letdown Flow | 0 to 110% design flow[1] | | | X | X |
| Sump Level in Spaces of Equipment Required for Safety | To corresponding level of safety equipment failure | | X | | X |
| RADWASTE SYSTEMS: | | | | | |
| High Level Radioactive Liquid Tank Level | Top to bottom | | | X | X |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% of design pressure[2] | | | | 0-150psig |
| VENTILATION SYSTEMS: | | | | | |
| Emergency Ventilation Damper Position | Open-closed status | X | X | X | X |
| Temperature of Space in Vicinity of Equipment Required for Safety | 30°F to 150°F | | | | 30°- 150°F |
| POWER SUPPLIES: | | | | | |
| Status of Class 1E Power Supplies and Systems | Voltages and currents | X | X | X | X |
| Status of Non-class 1E Power Supplies and Systems | Voltages and currents | X | X | X | X |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chipmen) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| **RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT:** | | | | | |
| Radiation Exposure Rates | $10^{-1}$ to $10^{4}$ R/hr for photons (permanently installed monitors) | X | X | X | X |
| **AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT:** | | | | | |
| Effluent Radioactivity -Noble Gases | (Normal plus accident range for noble gas) | | | | |
| -Containment | $10^{-7}$ to $10^{5}$ $\mu$Ci/cc Xe-133 calibration | X | X | X | X |
| -Secondary Containment | $10^{-7}$ to $10^{4}$ $\mu$Ci/cc Xe-133 calibration | | | | |
| -Auxiliary Building including buildings containing primary system gases, e.g. waste gas decay tank | $10^{-7}$ to $10^{3}$ $\mu$Ci/cc | | | | |
| -Other Release Points (including fuel handling area if separate from auxiliary building | $10^{-7}$ to $10^{2}$ $\mu$Ci/cc (permanently installed monitors) | | | | |
| Effluent Radioactivity -High Range Radiohalogens and Particulates | | | | | |
| Untreated Effluents | $10^{-3}$ to $10^{2}$ $\mu$Ci/cc | | | | |

TABLE 2 PWR VARIABLES continued

| Measured Variable | Range | NRC DRAFT 4-12-79 | NRC AGRS (Chirmer) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED: | | | | | |
| HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems | $10^{-3}$ to $10^{2}$ μCi/cc | | | | |
| HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems | $10^{-3}$ to 1 μCi/cc (permanently installed monitors) | | | | $10^{-3}$ to $10^{4}$ μCi/cc |
| Environs Radioactivity -High Range Exposure Rate | $10^{-3}$ to $10^{2}$ R/hr (60 keV to 3 MeV) (permanently installed monitors) | | | | X |
| Environs Radioactivity Radiohalogens and Particulates | $10^{-9}$ to $10^{-3}$ μCi/cc for both radiohalogens and particulates (permanently installed monitors) | | | | X |
| Plant and Environs Radioactivity (portable instruments) | Normal Range<br>0.1 to $10^{4}$ mR/hr photons | | | | X |
| | $10^{-9}$ to $10^{-4}$ μCi/cc particulates | | | | X |
| | $10^{-9}$ to $10^{-4}$ μCi/cc iodine | | | | X |
| | High Range<br>0.1 to $10^{4}$ R/hr photons | | | | -- |
| | 0.1 to $10^{4}$ rads/hr betas and low energy photons | | | | X |
| | 100-channel gamma-ray spectrometer | | | | X |

TABLE 2 PWR VARIABLES continued

| Measured Variables | Range | NRC DRAFT 4-12-79 | NRC ACRS (Chicman) | NRC List to ANS | ANS 4.5 |
|---|---|---|---|---|---|
| POST-ACCIDENT SAMPLING CAPABILITY: | | | | | |
| Primary Coolant Sumps Containment Air | As required based on Reg Guide 1.4 guidelines | | | | |
| POST-ACCIDENT ANALYSIS CAPABILITY (ONSITE): | 1. gamma-ray spectrum<br>2. pH<br>3. hydrogen<br>4. oxygen<br>5. boron | | | | |
| METEOROLOGY: | | | | | |
| Wind Direction | 0 to 360°(±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1mph) | | X | | X |
| Wind Speed | 0 to 30 mps (67mph) (±0.22mps(0.5mph) accuracy for wind speed less than 11mps (25 mph), with a starting threshold of less than 0.45 mps (1 mph) | | X | | X |
| Vertical Temperature Difference | -9°F to +9°F(±0.3°F accuracy per 164 foot intervals | | | | -5 to 5°F |
| Precipitation | Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value- 0.01" resolution | | | | |