POOR ORIGINAL

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JUL 12 1979

MEMORANDUM FOR: Robert B. Minogue, Director
Office of Standards Development

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: INSTRUMENTATION TO ASSESS NUCLEAR POWER PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT

One of the major lessons learned from the Three Mile Island accident is that better information needs to be provided to nuclear power plant operators to enable them to reliably assess what is taking place in the plant during an accident or transient situation so that they are better able to take remedial action. In addition to providing specific recommendations on instrumentation that should be required of licensees in the short term, the TMI Lessons Learned Task Force has strongly recommended that Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident, be revised on an expedited basis for early implementation. The purpose of this revision would be to incorporate the instruments already required by the Lessons Learned Task Force plus instruments that are determined to be necessary based on a more in-depth reanalysis of the past history of Regulatory Guide 1.97 in view of the experience of the TMI-2 accident. One important criterion that should guide the revision is the need to implement, as soon as practical, state of the art equipment in operating nuclear power plants to significantly increase the ability to follow the course of an accident. Long term instrument development matters should be deferred for further study pending results from longer term investigations and decisions flowing from TMI. We believe that a minimum set of basic instrumentation to follow an accident should be required of plants now in operation as well as those under construction on an expedited basis as soon as such a list is available.

During a meeting on July 3, 1979, between representatives from my office and your office, a course of action was discussed to accomplish an expeditious review and revision of Regulatory Guide 1.97. In accordance with the discussions during that meeting, I request that SD take the lead in this effort as follows:

a. An in-depth review of instrumentation needed to assess plant conditions during and following an accident should lead to a revision to R.G. 1.97 on an expedited basis; approximately two months to establish revised positions for review by the Regulatory Requirements Review Committee.

OFFICE						
SURNAME						
DATE						

b. Interest in providing assistance in this effort has been expressed by representatives of the national concensus standards committees and the Atomic Industrial Forum. Such assistance should be encouraged.

c. Ed Wenzinger, Chief, Reactor Systems Standards Branch, SD, will be in charge of this effort. In addition, SD will provide an engineer knowledgeable in the area of radiological monitoring.

d. NRR will assign Victor Benaroya of DSS and Leonard Soffer of DSE to assist in this effort.

If there is any problem in carrying out this effort, please let me know.

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: SLevine LGossick
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POOR ORIGINAL

7/11

all agreed in principal at Mtg w/RJM

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NUCLEAR RE ULATORY COMMISSION
WASHINGTON, D. C. 20555

DUPE 7904180060

MAR 2 8 1979

Generic Activity: A-34

MEMORANDUM FOR: Roger J. Mattson, Director
Division of Systems Safety

Roger S. Boyd, Director
Division of Project Management

Victor Stello, Jr., Director
Division of Operating Reactors

Edson G. Case, Chairman
Technical Activities Steering Committee

FROM: Richard C. DeYoung, Director
Division of Site Safety & Environmental Analysis

SUBJECT: DRAFT REPORT OF COMPLETION OF GENERIC ACTIVITY A-34

This report summarizes the results of staff efforts to develop guidance to facilitate implementation of Regulatory Guide 1.97, Revision 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident." As such this report documents completion of Generic Technical Activity A-34, "Instruments for Monitoring Radiation and Process Variables During Accidents." For reference, a copy of Task Action Plan A-34 is included as Appendix B to the draft report.

I request your comments and/or concurrence to issue this report as a NUREG. Please provide your comments to Fred Hebdon (x27066) by April 13, 1979.

Richard C. DeYoung, Director
Division of Site Safety and
Environmental Analysis

Enclosure:
As stated

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Draft

Report of Completion

of

Generic Activity A-34:

"Instruments for Monitoring Radiation and Process Variables During Accidents"

Division of Site Safety and
Environmental Analysis
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

1.0 INTRODUCTION

In December, 1975 the Staff issued for comment Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident." After reviewing the comments received the staff issued Revision 1 to this Regulatory Guide in August 1977. (A copy of Regulatory Guide 1.97, Revision 1 is provided in Appendix A).

The objective of Regulatory Guide 1.97 is to insure that during and following an accident, appropriate parameters and system functions are monitored in order that plant personnel will have sufficient information to take appropriate actions to restrict the courses and consequences of an accident. At the start of an accident, the operator cannot always determine what accident has occurred and therefore cannot always determine the appropriate response. For this reason, the reactor trip and certain safety actions (e.g. emergency core cooling actuation) are designed to be performed automatically during the initial stages of an accident. However, instrumentation is also necessary to provide information about plant parameters and system functioning that alerts the operator to conditions beyond those expected so that appropriate operator actions may be taken. The operator must have sufficient information available to: (1) determine the course of an accident; (2) make intelligent decisions about taking manual action; and (3) assist in determining what actions, if any, are needed to execute the plant emergency plan. It should be noted that it is not the intent of Regulatory Guide 1.97 that operators be encouraged to circumvent automatic features prematurely, but rather that they be adequately informed in order that they can take necessary planned and unplanned actions.

In August 1977, the staff issued Task Action Plan A-34, "Instruments for Monitoring Radiation and Process Variables During an Accident" (a copy of the most recent revision of the Task Action Plan is contained in Appendix B). The purpose of the Task Action Plan is to develop guidance for applicants, licensees and staff reviewers concerning implementation of Revision 1 of Regulatory Guide 1.97.

In the course of implementing the initial phase of the Task Action Plan, it became obvious that Regulatory Guide 1.97 included a few provisions which industry claimed to be impractical at the present time, and other provisions for which more definitive guidance was needed to define acceptable means of compliance. The primary issues in controversy are Positions C.1 and C.3 of the Regulatory Guide.

Position C.1 is intended to insure that the station design includes sufficient instrumentation to meet the objectives described in Position C.1 for each of the Design Basis Accidents normally analyzed by an applicant in Chapter 15 of a Safety Analysis Report.

Position C.3 describes specific instrumentation to be used if accident conditions degrade beyond those assumed in the FSAR. Various industry representatives expressed concern about the ranges of the instruments described in Position C.3 and the implication of monitoring for Class 9 accidents. This Position is not explicitly intended to monitor Class 9 accidents. Position C.3 is intended to provide assurance that even under conditions that degrade far beyond those that are assumed in the accident analyses, the operator will have usable instrumentation that will provide a basis for decision making. The operator must not be placed in a position where all his relevant instrumentation is off-scale. The ranges of the instruments described in Position C.3 are not based directly on accident scenarios but are based on engineering judgments of the admittedly extreme points beyond which the high probability of failure of important fission product barriers (e.g., reactor pressure vessel or containment structure) would make the need for instrumentation a moot point.

The remaining Positions in the Regulatory Guide describe the details of the design and qualification of the accident monitoring instrumentation and therefore do not pose the same type of implementation problems.

2.0 IMPLEMENTATION

During the months since issuance of Regulatory Guide 1.97 and Task Action Plan A-34, the staff and representatives of the nuclear industry have attempted to clarify the intent of the Regulatory Guide. Based on this work the staff has reached the following conclusions concerning implementation of Regulatory Guide 1.97 Revision 1.

1. The large amount of experience accumulated to date permits identification of those parameters that should be monitored to satisfy Position C.1. The list of parameters is provided as Appendix C. The staff will require that these parameters be monitored on all plants for which a construction permit application was docketed after Septembe ,0, 1977 (as per section D of Regulatory Guide 1.97 Revision 1). The accident monitoring instrumentation of plants for which a construction permit application was docketed prior to September 30, 1977 has been reviewed as part of the licensing process. Although the parameters monitored at specific plants may be different than those specified in Appendix C, the staff still believes that with the addition of the instruments described in Position C.3, existing accident monitoring equipment is acceptable. Therefore, the staff has concluded that the resources that would be required to backfit the instruments required to monitor the parameters listed in Appendix C would not be justified based on the benefits derived from having a standard set of accident monitoring instruments on all plants.

2. The staff concludes that technology currently exists to permit implementation of the instrumentation described in Positions C.3.a through C.3.c. Prior to issuance of Regulatory Guide 1.97 Revision 1 the staff did not require that accident monitoring instrumentation be provided with ranges extending beyond the conditions expected to result from Design Basis Accidents. For the reasons discussed in Section 1.0, the staff now believes that such instrumentation should be required on all plants. Therefore, the staff requires that the instrumentation described in Position C.3.a through C.3.c be implemented for reactor plant license applications and all plants licensed for construction or operation.
3. With respect to Position C.3.d, the staff is not certain that existing release rate monitoring technology is sufficient to permit adequate monitoring of the ranges of radioactivity release rates that might be encountered if, as assumed in Position C.3, conditions degrade beyond those expected to result from the Design Basis Accidents. Therefore, the staff will delay requiring implementation of Position C.3.d until studies of the capabilities of existing release rate monitoring technology can be undertaken.
4. It has been pointed out that it may not be feasible to qualify instrumentation to extreme conditions consistent with the instrument ranges described in Position C.3, particularly radiation levels inside containment of up to 10^8 rads/hour (Position C.3.b). The staff agrees that qualification of instrumentation located inside containment to such levels may not currently by possible.

However, the staff believes that all of the instrumentation described in Position C.3 can either be shielded or located outside the containment, where a less hostile environment would exist, and appropri:tely calibrated.

5. Position C.6 states that accident monitoring instrumentation should be de igned so that a single failure does not prevent the operator from accomplishing the objectives of Position C.1. However, it is the staff's position that redundant instrumentation is not required on each train of a system that has a redundant counterpart.
6. The staff worked closely with several applicants for construction permits and operating licenses, and with the Atomic Industrial Forum Ad Hoc Committee on Post Accident Monitoring Instrumentation. All of the concerns raised by the involved industry representatives have not been resolved to the satisfaction of all parties. However, the staff believes that sufficient guidance has been developed so that Task A-34 can be classified as complete. The staff will continue to work with the industry representatives in an attempt to resolve any minor issues that remain unresolved.

Task A-34

INSTRUMENTS FOR MONITORING RADIATION AND PROCESS VARIABLES DURING ACCIDENTS

Lead NRR Organization:	Division of Site Safety and Environmental Analysis (DSE)
Lead Supervisor:	Richard H. Vollmer A/D for Site Analysis, DSE
Task Manager:	Frederick J. Hebdon, Project Manager, Environmental Projects Branch 1, DSE
Applicability:	All Reactor Types
Projected Completion Date:	November 1978

APPENDIX B

1. DESCRIPTION OF PROBLEM

To develop criteria and guidelines to be used by applicants, licensees and staff reviewers to support implementation of Regulatory Guide 1.97, Revision 1 (Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident).

Such criteria and guidelines would provide specific guidance on functional and operational capabilities required of the various classes of instruments, including inplant and explant instruments. Where such guidance cannot be provided, the rationale to be applied to derive requirements for specific situations will be provided.

2. PLAN FOR PROBLEM RESOLUTION

A. Detailed guidance and acceptance criteria concerning implementation of Regulatory Guide 1.97 has not yet been developed. Therefore, the members of this Task Group will answer questions that arise before and during the development of the required proposals for implementation of Regulatory Guide 1.97 for the lead plants described below. In this way, the Task Group will develop the necessary guidance as it is needed by the lead plant applicants. The Task Group will also be responsible for the review of submittals made by the lead plant applicants.

B. There are two aspects of the implementation of Regulatory Guide 1.97, Revision 1 (Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident) that must be considered.

(1) Position C.3 of RG 1.97 requires the installation of specific instrumentation to follow the course of an accident (IFCA). The staff has determined that this requirement should be satisfied in as timely a manner as possible. The Task Group established by this Task Action Plan will identify lead plants (at least one BWR and one PWR) for implementation of Position C.3, will answer questions raised by the lead plant applicants, and will assume responsiblity for the review of the proposals for implementation of Position C.3 that are submitted. Based on the experience gained during this review, the Task Group will prepare uniform review procedures and acceptance criteria to be used by the staff for the review of subsequent implementation proposals.

(2) Full implementation of RG 1.97 requires the applicant/licensee to prepare a Safety Analysis which is reviewed by the staff. Lead plants (at least one BWR and one PWR) for full implementation of RG 1.97 will be designated. The Task Group established by this Task Action Plan will assist the lead plant applicants in the development of the required Safety Analyses by answering questions from the applicants. The Task Group will review the Safety Analyses when they are submitted. Based on the experience gained during the development and review of the Safety Analyses for the lead plants, the Task Group will prepare guidance to assist other applicants/licensees in the development of the required Safety Analysis and acceptance criteria to be used by the staff to review the Safety Analyses submitted.

C. Description of the End Product of Task Group

(1) A letter to all applicants and licensees containing guidance to facilitate the preparation of Safety Analyses required by RG 1.97.

(2) Revision of various Standard Review Plans to provide for the uniform review of required Safety Analyses and Proposals for Implementation of Position C.3.

(3) Recommendation for revision of RG 1.70, Standard Format and Content of SAR's for Nuclear Power Plants.

(4) Recommendations for confirmatory research as required.

(5) Recommendations for revisions to RG 1.97.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

As described in Sections 1 and 2, the issue addressed by this task is the timely development of criteria and guidelines to support full implementation of Reg. Guide 1.97, Revision 1 for CP's, OL's and Operating Reactors. Full implementation of Reg. Guide 1.97, Revision 1 requires the applicant/licensee to prepare a Safety Analysis (of instruments to follow the course of an accident which are part of this task as opposed to instruments to prevent an accident which are not) which is to be reviewed by the staff. This task will provide guidance to applicants for preparation of the Safety Analysis report and criteria and analyses by which the staff will review the report.

The current staff review process assures that the likelihood of serious accidents is extremely low. Implementation of the defense-in-depth concept and the single failure criterion assure that there is no undue risk to the health and safety to the public. There is, however, a residuum of risk from accidents which are more severe than those evaluated in the applicant's Safety Analysis Report and reported on in the staff's Safety Evaluation Report. This residuum of risk is small when compared to other risks in society and as such, specific designs to accommodate accident conditions contributing to these risks is not required. The staff has, however, determined that it is prudent to provide additional capability for plant operators to identify accident conditions which could lead to significant consequences. Full implementation of the provisions of Regulatory Guide 1.97 Revision 1 will provide additional assurance that the operator will be able to identify the need for and execute accident mitigation procedures for design basis accidents and be able to identify and act to rectify accident conditions which have been degraded beyond the design basis. The low level of the residual risk for current designs presents no undue risk to the health and safety of the public.

3. NRR TECHNICAL ORGANIZATIONS INVOLVED

These branches will carry out their responsibilities through participation on the Task Group.

A. Accident Analysis Branch (DSE) - review the Safety Analyses required by RG 1.97 for the lead plants to ensure that variations in plant variables are adequately defined, from a consequences viewpoint, for the Design Basis Accidents analyzed. This review will also include evaluation of operator interaction (e.g., procedures, actions, timing) for utilizing instrumentation to follow the course of an accident (IFCA) to assess and minimize risk. Develop guidance for applicants/licensees and uniform review procedures for the staff to support the implementation of RG 1.97 on other plants. Review the plans for implementation of Position C.3 for lead plants and develop uniform review procedures for the staff to use to review implementation proposals for other plants. (Manpower Requirements: 1 reviewer, 2MM per reviewer.)

B. Reactor Systems Branch (DSS), Containment Systems Branch (DSS), Auxiliary Systems Branch (DSS), Power Systems Branch (DSS)

Review the Safety Analyses for the lead plants to ensure that significant process variables required to monitor the course of Design Basis Accidents, from a systems performance viewpoint, are identified. This review will also include evaluation of operator interactions (e.g., procedures, actions, timings) for utilizing IFCA to optimize system performance. Develop guidance for applicants/licensees and uniform review procedures for the staff to use to implement RG 1.97 on other plants. (Manpower requirements: 1 reviewer per branch, 3MM per reviewer in RSB, 1MM per reviewer in CSB, and PSB.)

C. Radiological Assessment Branch (DSE) and Effluent Treatment Systems Branch (DSE) - develop criteria for application of inplant and explant radioactivity monitoring systems to follow the course of an accident during various accident situations and accident scenarios. Review the Safety Analyses for the lead plants to ensure that plant radiation sources are adequately defined and that radiation monitoring is adequate from the viewpoint of protection of the health and safety of utility staff personnel, of emergency program personnel and of the public outside the immediate plant environs. (Manpower requirements: 1 reviewer, 2 MM per reviewer for RAB and 1 reviewer, 1 MM per reviewer for ETSB).

D. Instrumentation and Control Systems Branch (DSS) - review the Safety Analyses for the lead plants to ensure that IFCA is appropriately designed, will remain operable as required, and will accurately represent the information required by the operator. This review will include consideration of maintenance and testing of instrumentation. Develop guidance for applicants/licensees and review procedures for the staff to use to implement RG 1.97 on other plants. Review the plans for implementation of Position C.3 for lead plants and develop uniform review procedures for the staff to support the review of implementation proposals for other plants. (Manpower Requirements: 1 reviewer, 2MM per reviewer.)

E. Operator Licensing Branch (DPM) - assist in evaluating operator interactions and expected operator responses to identify the instrumentation required and the procedures to be followed to deal with Design Basis Accidents. Develop guidance for applicants/licensees and uniform review procedures for the staff to support implementation of RG 1.97 on other plants. (Manpower Requirements: 1 reviewer, 1MM per reviewer.)

F. Emergency Planning Branch (DPM) - review the Safety Analyses for lead plants and the applicant's Emergency Plan to ensure that the operator will be supplied with the information needed to permit him to provide authorities responsible for implementation of Emergency Plan with accurate and timely recommendations concerning implementation of all or part of the plan. Develop guidance for applicants/licensees and uniform review procedures for the staff to support implementation of RG 1.97 on other plants. Review the plan of Position C.3 for lead plants and develop uniform review procedures for the staff to support the review of implementation proposals for other plants. (Manpower Requirements: 1 reviewer, 1MM per reviewer.)

G. Environmental Projects Branch 1 (DSE) - Provide a Task Manager to serve in the principal management function for the project. (Manpower Requirements: 1 project manager, 3MM manager.)

H. Operating Technology (DOR) - Review and comment on materials developed by the Task Group. Adapt the criteria and guidance developed by the Task Group for use by reviewers and licensees of operating reactors. [Manpower Requirements: 1 reviewer per branch (4 branches), 1 MM per reviewer.]

I. Other Branches in NRR may be called upon to provide technical support to the Task Group as needed on a consultation basis. (Manpower Requirements: Total 1 MM.)

5. TECHNICAL ASSISTANCE FUNDS AND CONFIRMATORY RESEARCH FUNDING REQUIRED

It is not presently anticipated that technical assistance funding or confirmatory research funding will be required to directly support this Task Group. Two projects (described below) may produce data that will support the activities of this Task Group.

A. DOR has an existing technical assistance contract with BNL to evaluate certain operating plants to determine the capability of existing effluent radiation monitors to measure radioactivity releases through anticipated release paths from postulated accidents. The funding level for this program is $25K for FY 1977 and FY 1978.

B. DSE has an existing technical assistance contract with Allied Chemical Company (INEL) to develop bases for the specification of gaseous effluent accident monitoring instrumentation. The funding level for this program is $40K for FY 1977.

6. INTERACTION WITH OUTSIDE ORGANIZATIONS

The Task Group will maintain close contact with applicants for the lead plants.

7. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

Office of Standards Development - Assist in the development of subsequent revisions of RG 1.97 and other Regulatory Guides based on experience gained during the review of the lead plants.

8. POTENTIAL PROBLEMS

Based on preliminary studies, as exemplified in BNWL-1635, it is anticipated that many plant evaluations, particularly those for operating plants, will show the need for monitoring equipment not commercially available and, therefore, a lead time of six months to two years may be necessary for development, procurement, and installation of monitoring equipment.

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

Parameter

Containment pressure

Hot leg flow (PWR)

Cold leg flow (PWR)

Level in steam generator

Main steamline flow rate

Pressure of reactor coolant

Pressurizer level (PWR)

Radiation level in condenser air ejector

Steam-generator pressure (PWR)

Temperature of reactor coolant

Position of Valves in Vital Systems

Component cooling water system Flow

Containment cooling fan flow

Containment spray flow

Containment sump and suppression pool level

Control rod position indicators

Emergency cooling water storage tank level

Emergency filter train operation

Emergency ventilation system(s) damper positions

Injection flow

Power (Neutron flux)

Residual heat removal flow

Appendix C

Parameter

Safety injection flow

Status of power supplies

Ultimate heat sink temperature and level

Area radiation levels in auxiliary buildings

Boron concentration and/or flow (PWR)

Containment temperature

Hydrogen concentration in containment

Radiation level in containment

Radiation level in main steamline (BWR)

Reactor vessel coolant level

Temperature of space in vicinity of vital equipment

Activity levels in surface and ground water

Activity release rate from principle plant vents and discharge points

Wind direction, speed and vertical temperature difference

Environmental Radiation Levels

Revision 1
August 1977

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

dupe of 7907100354

REGULATORY GUIDE 1.97

INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT*

A. INTRODUCTION

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

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50 includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents.

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

This guide describes a method acceptable to the NRC staff for complying with the Commission's requirements to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

Monitored variables and systems are used by the operator in accident surveillance to (1) assist in determining the nature of an accident; (2) determine whether the reactor trip and engineered-safety-feature systems are functioning properly; (3) determine whether the plant is responding properly to the safety measures in operation; (4) provide information to the operator that will enable him to determine the potential for breaching the barriers to radioactivity release; (5) furnish data for deciding on the need to take manual action if an engineered safety feature malfunctions or the plant is not responding effectively to the safety systems in operation; (6) allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of the impending threat; and (7) aid in determining the cause and consequence of the event for postaccident investigation.

At the start of an accident, the operator cannot always determine immediately what accident has occurred or is occurring and therefore cannot always determine the appropriate response. For this reason, the reactor trip and certain safety actions (e.g., emergency core cooling actuation, containment isolation, or depressurization) are designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant parameters required to enable the operation of manually initiated safety-related systems and other appropriate operator actions.

Examples of serious events that threaten safety if conditions degrade beyond those assumed in the Final Safety Analysis Report are loss-of-coolant accidents (LOCAs), reactivity excursions, and radioactivity releases. Such events require that the operator understand, in a short time period, the state of readiness of engineered safety features and their potential for being challenged by an accident in progress.

* Lines indicate substantive changes from previous issue.

APPENDIX A

To determine the important variables and the items whose values or status are needed by the operator and, therefore, the monitoring instrumentation needed by the operator, a study (Ref. 1) was made of a range of postulated accidents. The study concluded that the following capabilities are most important to ensuring that the power plant poses no threat to public safety after an accident: reactor shutdown, core cooling, containment isolation, and the maintenance of containment pressure control, primary system pressure control, and a heat transfer path from the core to a heat sink. These vital capabilities are designed to preserve the integrity of the barriers to radioactivity release (i.e., the fuel cladding, reactor coolant boundary, and containment).

It is essential that the required instrumentation be capable of surviving the accident environment in which it is located for the length of time its function is required. It could therefore either be designed to withstand the accident environment or be protected by a local artificial environment. If the environment surrounding an instrument component is the same for accident and normal operating conditions (e.g., the instrumentation components in the main control room), the instrumentation components need no special environmental capability.

It is important that accident-monitoring instrumentation components and their mounts that cannot be located in other than non-Seismic Category I buildings be conservatively designed for the intended service.

Parameters selected for accident monitoring can be selected so as to permit relatively few instruments to provide the essential information needed by the operator for postaccident monitoring. Further, it is prudent that a limited number of those parameters (e.g., containment pressure) be monitored by instruments qualified to more stringent environmental requirements and with ranges that extend to the maximum values that the selected parameters can attain under worst-case conditions; for example, a range for the containment pressure monitor extending beyond the design pressure of the containment.

Normal power plant instrumentation remaining functional for all accident conditions can provide indication, records, and (with certain types of instruments) time-history responses for many parameters important to following the course of the accident. Therefore, it is prudent to select the required accident-monitoring instrumentation from the normal power plant instrumentation. Since some accidents impose severe operating requirements on instrumentation components, it may be necessary to upgrade some instrumentation components to withstand the more severe operating conditions and to measure greater variations of monitored variables that may be associated with the accident if they are to be used for both accident and normal operation. However, it is essential that instrumentation so upgraded does not compromise the accuracy and sensitivity required for normal operation.

It should be noted that in the safety analysis many parameters may be identified that will provide desirable, but less essential, information for the operator. Any instrumentation used to measure these less essential (i.e., "backup") parameters is outside the scope of this guide.

C. REGULATORY POSITION

1. For the postulated accidents listed in Chapter 15 of Regulatory Guide 1.70 (Ref. 2), the applicant should perform detailed safety analyses necessary to determine the parameters to be measured and the instrument ranges, responses, accuracies, and length of time required to provide the operator with the information necessary to:

a. Assist in determining the nature of an accident,

b. Determine whether the reactor trip and engineered-safety-feature systems are functioning properly,

c. Determine whether the plant is responding properly to the safety measures in operation,

d. Determine the potential for breaching the barriers to radioactivity release,

e. Decide on the need to take manual action if an engineered safety feature malfunctions or the plant is not responding effectively to the safety systems in operation, and

f. Allow for early indication of necessary action to protect the public and for an estimate of the magnitude of the impending threat.

The guidelines in Reference 1, along with the guidelines in Reference 3 dealing with monitoring inside the power plant, may be used to make such analyses.

2. The instrumentation necessary to provide the information noted in regulatory position 1 should be specified along with justification to show that the instrumentation is adequate to provide the operator with the necessary information. The safety analyses should provide the information necessary to select the appropriate type of accident-monitoring instrument; to specify the range, accuracy, transient response, environmental and seismic qualifications, and insensitivity to variations of energy supply; and to specify the method of recording, when recording is deemed necessary.

3. A limited number of additional accident-monitoring instruments should have ranges that extend to the maximum values that selected parameters can attain under worst-case conditions, and the in-

strumentation components should be qualified to withstand the higher level of environmental conditions in which they will be required to function. These parameters and associated maximum values to be measured by the instruments should include, but not necessarily be limited to, the following:

a. Containment pressure: 3 times design pressure for concrete; 4 times design pressure for steel.

b. Radiation level inside containment: 10^8 rads per hour.

c. Reactor coolant pressure: 3 times design pressure.

d. Plant radioactivity release rate through identifiable release points: (plant dependent) (range dependent on maximum release rate postulated for a given release point).

4. The accident-monitoring instrumentation should be qualified in accordance with Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants."

Instrumentation that is Seismic Category I, as defined by Regulatory Guide 1.29, "Seismic Design Classification," should continue to function within the required accuracy following, but not necessarily during, a safe shutdown earthquake.

Instrumentation components and their mounts that cannot be located in other than non-Seismic Category I buildings need not meet Seismic Category I criteria.

5. Those parameters selected for accident-monitoring instrumentation that provide transient or trend information necessary for the operator to perform his role should be recorded. Records of parameters that provide information related to the determination of radioactivity release rates and total radioactivity releases should be considered necessary.

6. The accident-monitoring instrumentation should be designed so that a single failure does not prevent the operator from accomplishing the objectives of regulatory position 1

NOTE: "Single failure" includes such events as the shorting or opencircuiting of interconnecting signal or power cables. It also includes single credible malfunctions or events that cause a number of consequential component, module, or channel failures. For example, the overheating of an amplifier module would be a "single failure" even though several transistor failures might result. Mechanical damage to a mode switch would be a "single failure" although several channels might become involved.

7. The accident-monitoring instrumentation channels that are redundant should be electrically independent, energized from station Class IE power, and physically separated, in accordance with Regulatory Guide 1.75, "Physical Independence of Electric Systems."

8. To the extent practical, accident-monitoring instrumentation inputs should be from sensors that directly measure the desired variables.

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

10. The accident-monitoring instrumentation should be specifically identified on control panels so that the operator can easily discern that they are intended for use under accident conditions.

11. Any equipment that is used for both accident monitoring and nonsafety functions should be classified as part of the accident-monitoring instrumentation. The transmission of signals from accident-monitoring equipment for nonsafety system use should be through isolation devices that are classified as part of the accident-monitoring instrumentation and that meet the provisions of the document.

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

a. By perturbing the monitored variable;

b. By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or

c. By cross-checking between channels that bear a known relationship to each other and that have readouts available.

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

EXCEPTION: "One-out-of-two" systems are permitted to violate the single-failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated. For example, the bypass time interval required for a test, calibration, or maintenance operation could be shown to be so short that the probability of failure of the active channel would be commensurate with the probability of failure of the "one-out-of-two" systems during its normal interval between tests.

14. Whenever means for bypassing channels are included in the design, the design should permit administrative control of the access to such bypass means.

15. The design should permit administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

16. The accident-monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications confusing to the operator.

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

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with the specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals for construction permit applications docketed after September 30, 1977.

REFERENCES

1. Battelle-Columbus Laboratories, "Monitoring Post-Accident Conditions in Power Reactors," BMI-X-647, April 9, 1973.
2. U.S. Nuclear Regulatory Commission, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," NUREG-75/094, Regulatory Guide 1.70, Revision 2, September 1975.
3. BNWL-1635, "Technological Considerations in Emergency Instrumentation Preparedness," May 1972.