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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.

To determine the important variables and the systems 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.