NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

REVIEW RESPONSIBILITIES

Primary - Instrumentation and Control Systems Branch (ICSB)

Secondary - Auxiliary Systems Branch (ASB)
Reactor Systems Branch (RSB)
Quality Assurance Branch (QAB)
Power Systems Branch (PSB)

I. AREAS OF REVIEW

The areas reviewed in this section of the applicant's safety analysis report (SAR) include those control systems identified by the applicant as being nonsafety-related. These control systems may include the primary system pressure, temperature and water level controls, and feedwater controls. The intent of the review is to assure that failures of these controls would not impair the protection system capability in any significant manner. Since the control systems of interest under this SRP section may vary from plant to plant depending upon individual designs, the applicant should identify all such systems and provide analyses to support their classification as nonsafety-related control systems.

The ICSB will review the following aspects of the nonsafety-related control systems: the circuit-to-circuit failure modes of a single nonsafety-related control system and their effect on the protection system, and gross failure modes of nonsafety-related control systems and their functional effect on the protection system.

The ASB, PSB and RSB provide assistance in verifying that all control [illegible] have been identified and that the input signal parameters for the control system [illegible] rect. The RSB determines that the control systems identified in this SAR section [illegible] required for safety and that no credit is taken in the plant accident analyses for the control systems so identified.

The PSB reviews the turbine generator control and overspeed protection systems set forth in SRP Section 10.2.

The QAB verifies that the quality assurance program implemented for control system components, where necessary, is adequate.

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II. ACCEPTANCE CRITERIA

The control systems not required for safety are acceptable if failures of control system components or total systems would not significantly affect the ability of plant safety systems to function as required, or cause plant conditions more severe than those for which the plant safety systems are designed.

Table 7-1 lists those General Design Criteria (GDC) of Appendix A to 10 CFR Part 50, and standards of the Institute of Electrical and Electronic Engineers (IEEE), that are used as references in arriving at this conclusion. General Design Criteria 13 and 24 and IEEE Std 279, Section 4.7, are of special importance among these references.

1. Conformance with General Design Criterion 13 for Instrumentation and Control Requirements. Instrumentation should be provided to monitor variables and systems over their anticipated ranges for normal operation and for anticipated operational occurrences as appropriate to minimize challenges to safety systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

2. Conformance with General Design Criteria 24 for Separation of Control Systems from Protection Systems.
The protection system shall be separated from control systems to the extent that failure of any single control system component or channel which is common to control and protection systems shall not violate the reliability, redundancy, and independence requirements of the protection system. The interconnections between the protection and control system shall be limited so as to assure that safety is not significantly impaired.

3. Conformance to IEEE Std 279, Section 4.7, for Control and Protection System Interaction. The direct circuit-to-circuit and functional interactions between control and protection systems for single random or multiple failures in the control system shall not prevent the protection system channel from meeting the minimum performance requirements specified in the design bases.

For those areas of review identified in subsection I of this SRP section as being the responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections corresponding to those branches.

III. REVIEW PROCEDURES

1. The objectives in the review are:

   a. To establish that control systems identified as being nonsafety-related, which, depending upon plant design, may include the primary system pressure, temperature, and feedwater controls, and steam generator water level controls, are, in fact, not required for plant safety.

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b. To verify that no credit is taken for the operability of these control systems in the plant accident analyses in Chapter 15 of the SAR.

c. To assure that failures of these control systems would not impair the capability of the protection system in any significant manner or cause plant conditions more severe than those for which the plant safety systems are designed.

d. To establish that control system designs meet applicable requirements of the General Design Criteria and industry standards with regard to independence between control and protection functions.

2. In the construction permit (CP) review the descriptive information, including the design bases and preliminary analyses, are reviewed to determine that there is reasonable assurance that the final design will meet these objectives. The RSB, PSB and ASB identify the plant systems whose control system designs are to be reviewed to verify that no credit is taken for their operability in the plant accident analyses. ICSB reviews the descriptive information provided for those systems at the construction permit stage to assure that control and protective functions are adequately separated, to assess the effects of control system failures, and to verify that commitments are made that such failures will be included in the plant safety design bases.

3. At the operating license (OL) stage, the objectives described in item 1., above, are verified during the review of control system schematics. At the operating license stage, ICSB reviews electrical schematic drawings for these control systems as necessary to assure that adequate attention has been given to the separation of control and protective functions and to possible effects of failures of these systems. The review includes interactions between control systems and effects on plant safety systems due to control system malfunctions or failures.

4. Upon request for the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

5. A typical review procedure for pressurized water reactor (PWR) primary and secondary control system functions follows:

   a. The primary system pressure is maintained within specified limits by the use of pressurizer heaters and spray valves. The primary pressure control system description and schematics are reviewed:

      (1) To confirm that the system will maintain the primary coolant pressures within prescribed limits for normal and transient operating conditions.
      (2) To determine the effects of loss of power to the pressurizer heaters and spray valves.
      (3) To determine the effects of loss of air to any pneumatically-operated valves in the spray system.

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Assistance as needed is obtained from the RSB in evaluating these items.

b. To meet the requirements of General Design Criterion 24 and Section 4.7 of IEEE Std 279 on control system interactions with the protection system, loss of primary pressure control function is analyzed. Assistance is obtained from RSB in establishing the sequence of events that would follow. The evaluation should show that failure of the primary pressure control system would not significantly degrade the capability of the protection system. Also, the reviewer determines that where a random failure in the pressure control system can result in a plant condition requiring protective action and can also prevent proper action of a protection channel designed to protect against the condition, the remaining redundant channels will provide the protective action even when degraded by another random failure.

c. The system description and control schematics of the feedwater regulating system are reviewed to idenfity failure modes of the system components. Assistance is obtained from the RSB, PSB and ASB in identifying the control function parameters. The system actions are established for loss of air to the feedwater control valves and malfunction in the feedwater heater bypass valves. The reviewer should verify that manual override of the automatic control is designed into the system.

d. The reviewer evaluates the effects of multiple failures in control systems resulting from single events. Failures in the secondary system water level (i.e., feedwater flow and steam generator water level) controls are analyzed along with failure in the primary coolant pressure control, where a single event can cause these multiple failures. With the assistance from the RSB and ASB, the reviewer determines that control function failures of both primary pressure and secondary water level controls would not prevent the minimum required number of reactor protection system channels from tripping the reactor.

In certain instances, it will be the reviewer's judgment that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a nonuniform placement of emphasis are the introduction of new design features or the utilization of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

At the construction permit stage, it should be established that the information and commitments documented in the preliminary safety analysis report (PSAR) provide reasonable assurance that the final designs of nonsafety-related control systems will conform with the intent of this SRP section.

At the operating license stage, sufficient design detail for these control systems is reviewed to determine adequate conformance. Exceptions to the acceptance basis given in subsection II are identified, with a statement as to how these exceptions provide a conservative basis for engineering design of the affected control systems.

The reviewer verifies that sufficient information has been submitted and the review supports conclusions of the following type, to be included in the staff's evaluation report:

> "The staff has reviewed the controls for systems not required for safety, to determine the affects of failures or malfunctions of these controls on the reactor protection system and other plant safety-related systems. The nonsafety-related control systems are (identify control systems so designated). We conclude that failures or malfunctions of these controls would not be expected to degrade the capabilities of plant safety systems in any significant degree, or to lead to plant conditions more severe than those for which the safety systems are designed."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems."

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APPENDIX 7-A BRANCH TECHNICAL POSITIONS (ICSB)

The ICSB Branch Technical Positions (BTPs) represent guidelines intended to supplement the acceptance criteria established in Commission regulations and regulatory guides, and in applicable IEEE standards. The BTPs originate in technical problems or questions of interpretation that arise in the detailed reviews of plant designs. The staff must make a judgment in each such case, in order to complete its review of the particular application. Where the same technical problem or question of interpretation arises in several cases, the staff's judgment on the point at issue is formalized in a BTP. The BTP is primarily an instruction to staff reviewers that outlines an acceptable approach to the particular issue and ensures a uniform treatment of the issue by staff reviewers. The approaches taken in the BTPs, like the recommendations of regulatory guides, are not mandatory, but do provide defined, acceptable, and immediate solutions to some of the technical problems and questions of interpretation that arise in the review process. In some instances, regulatory guides may be developed from BTPs after a sufficient experience in their use has accumulated.

All ICSB BTPs applicable to the SRP sections in Chapter 7 have been collected in this Appendix for convenience. Other ICSB BTPs applicable to Chapter 8 are presented in SRP Appendix 8-A. When another branch or division is assigned review responsibility for a BTP, that branch or division identified parenthetically as part of the BTP designation.

BTP ICSB	Branch Technical Positions of the Instrumentation and Control Systems Branch
1. (DOR)	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors.
3.	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System.
4. (PSB)	Requirements of Motor-Operated Valves in the ECCS Accumulator Lines.
5.	Scram Breaker Test Requirements - Technical Specifications.
9.	Definition and Use of "Channel Calibration" - Technical Specifications.

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12. Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service.

13. Design Criteria for Auxiliary Feedwater Systems.

14. Spurious Withdrawals of Single Control Rods in Pressurized Water Reactor.

15. (PSB) Reactor Coolant Pump Breaker Qualification (attached to SRP Appendix 8-A).

16. Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors.

18. (PSB) Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves (attached to SRP Appendix 8-A).

19. Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems.

20. Design of Instrumentation and Controls Provided to Accomplish Change-over from Injection to Recirculation Mode.

21. Guidance for Application of Regulatory Guide 1.47.

22. Guidance for Application of Regulatory Guide 1.22.

25. Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole.

26. Requirements for Reactor Protection System Anticipatory Trips.

BRANCH TECHNICAL POSITION ICSB 1 (DOR)

BACKFITTING OF THE PROTECTION AND EMERGENCY POWER SYSTEMS OF NUCLEAR REACTORS

A. BACKGROUND

The acceptance criteria used by the staff in the evaluation of protection and emergency power systems undergo improvement from time to time. With each change it is necessary to determine whether previously approved designs should be modified (backfitted) to meet the revised criteria. Ths determination is made on the basis of whether a significant incremental increase in safety of the plant would be obtained that would justify the various difficulties of the change.

The actions which raise the question of possible backfitting are:

1. Application for a full-term operating license for plants now operating with a provisional operating license.

2. Evaluation of a significant plant modification proposed by the staff or the licensee.

3. Application for a full-term operating license for plants now operating under DOD 91-B exemptions.

B. BRANCH TECHNICAL POSITION

For cases falling in the categories 1-3 in (A) above, the following apply;

1. Instrumentation and electric equipment essential to safety which must function in an accident environment should be analyzed or tested to demonstrated this capability.

2. Protection circuits essential to safety should meet the single failure criterion of Section 4.2 of IEEE 279.

3. Where d-c power is required for safety, redundant d-c sources should be provided and the d-c circuits should meet the single failure criterion.

4. For reactor plants supplying electric power to electric utility grids, redundant sources of onsite a-c power should be provided and the a-c circuits should meet the single failure criterion. This aspect of the design of research and test reactors should be evaluated on an individual case basis.

C. REFERENCES

None

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BRANCH TECHNICAL POSITION ICSB 3

ISOLATION OF LOW PRESSURE SYSTEMS FROM THE HIGH PRESSURE REACTOR COOLANT SYSTEM

A. BACKGROUND

During [illegible]al and emergency conditions, it is necessary to keep low pressure systems that are connected to the high pressure reactor coolant system properly isolated in order to avoid damage by overpressurization or the potential for loss of integrity of the low pressure system and possible radioactive releases. There have been a number of recommendations for accomplishing this aim. Until a more definitive guide is published, the criteria in Part B, below, provide an adequate and acceptable design solution for this concern.

B. BRANCH TECHNICAL POSITION

The following measures should be incorporated in designs of the interfaces between low pressure systems and the high pressure reactor coolant system:

1. At least two valves in series should be provided to isolate any subsystem whenever the primary system pressure is above the pressure rating of the subsystem.

2. For system interfaces where both valves are motor-operated, the valves should have independent and diverse interlocks to prevent both from opening unless the primary system pressure is below the subsystem design pressure. Also, the valve operators should receive a signal to close automatically whenever the primary system pressure exceeds the subsystem design pressure.

3. For those system interfaces where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent the valve from opening whenever the primary pressure is above the subsystem design pressure, and to close automatically whenever the primary system pressure exceeds the subsystem design pressure.

4. Suitable valve position indication should be provided in the control room for the interface valves.

5. For those interfaces where the subsystem is required for ECCS operation, the above recommendations need not be implemented. System interfaces of this type should be evaluated on an individual case basis.

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 4 (PSB)

REQUIREMENTS OF MOTOR-OPERATED VALVES IN THE ECCS ACCUMULATOR LINES

A. BACKGROUND

For many postulated loss-of-coolant accidents, the performance of the emergency core cooling system (ECCS) in pressurized water reactor plants depends upon proper functioning of the safety injection tanks (also referred to as "accumulators" or "flooding tanks" in some applications). In these plants, a motor-operated isolation valve (MOIV) and two check valves are provided in series between each safety injection tank and the reactor coolant (primary) system.

The MOIVs must be considered to be "operating bypasses" because, when closed, they prevent the safety injection tanks from performing the intended protective function. IEEE Std 279 has a requirement for "operating bypasses" which states that the bypasses of a protective function will be removed automatically whenever permissive conditions are not met. This Branch Technical Position provides specific guidance in meeting the intent of IEEE Std 279 for safety injection tank MOIVs.

It should be noted that BTP ICSB 18 (PSB), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," also applies to these isolation valves and should be used in conjunction with this position.

B. BRANCH TECHNICAL POSITION

The following features should be incorporated in the design of MOIV systems for safety injection tanks to meet the intent of IEEE Std 279.

1. Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications), or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator.

2. Visual indication in the control room of the open or closed status of the valve.

3. An audible and visual alarm, independent of item (2), above, that is actuated by a sensor on the valve when the valve is not in the fully-open position.

4. Utilization of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the technical specifications).

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C. REFERENCES

1. Arkansas 1, Unit 1, Safety Evaluation Report, January 23, 1973.

2. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

3. BTP ICSB 18 (PSB), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves."

BRANCH TECHNICAL POSITION ICSB 5

SCRAM BREAKER TEST REQUIREMENTS - TECHNICAL SPECIFICATIONS

A. BACKGROUND

There have been some inconsistencies in the description of scram circuit test procedures in FSARs and technical specifications requirements. Some FSARs for plants with Westinghouse reactors describe the scram circuit test procedures and include a position for testing the scram breakers, but there are no provisions for doing so in the proposed technical specifications. It is the purpose of this Branch Technical Position to establish a uniform practice in this matter.

B. BRANCH TECHNICAL POSITION

The requirement that control rod drive trip breakers be tested monthly should be included in all plant technical specifications issued. For a model, refer to the Oconee Technical Specifications page 4.1-4, Table 4.1-1, item 2.

C. REFERENCES

None

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BRANCH TECHNICAL POSITION ICSB 9

DEFINITION AND USE OF "CHANNEL CALIBRATION" - TECHNICAL SPECIFICATIONS

A. BACKGROUND

In several PWR technical specifications, the term "channel calibration" was used to describe a "daily adjusatment" for amplifier gain of the nuclear instrumentation power range channels. This adjustment was performed to maintain agreement between the indicated reactor nuclear power level and the reactor thermal power calculation. This adjustment is not considered by the staff to be a channel calibration. A calibration procedure performed on a monthly basis requires the following:

1. Performance of a functional test using a simulated signal to verify bistable action (protective trips including rod block trips and permissive interlocks) on a monthly basis.

2. Calibration of the upper and lower chambers of each flux channel for axial offset utilizing the in-core detectors on a calendar quarter basis.

3. Performance of a functional test using a simulated signal to verify positive and negative rate bistable action on a monthly basis.

Performance of a total system response time is required during each refueling outage.

B. BRANCH TECHNICAL POSITION

The "daily adjustment," which does not fulfill the intent or requirements of a calibration procedure, should remain as a daily requirement but be deleted from the "channel calibration" category in the technical specifications.

C. REFERENCES

None

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BRANCH TECHNICAL POSITION ICSB 12

PROTECTION SYSTEM TRIP POINT CHANGES FOR OPERATION WITH REACTOR COOLANT PUMPS OUT OF SERVICE

A. BACKGROUND

For the past several years, including a time prior to the development of IEEE Std 279, the staff has required automatic adjustment to more restrictive settings of trips affecting reactor safety by means of circuits satisfying the single failure criterion. The basis for this requirement is that the function can be accomplished more reliably by automatic circuitry than by a human operator. This design practice, which has also been adopted independently by the national laboratories and by much of industry, served as the basis for paragraph 4.15, "Multiple Set Points," of IEEE Std 279.

More recently, all applicants have stated that their protection systems were designed to meet IEEE Std 279. Paragraph 4.15 of IEEE Std 279 specified that where a mode of reactor operation requires a more restrictive set point, the means for ensuring use of the more restrictive set point shall be positive and must meet the other requirements of IEEE Std 279. A number of designs have been proposed and accepted which reliably and simply satisfy this requirement. During the review of some applications, however, certain design deficiencies have been found. The purpose of this position is to provide additional guidance on the application of Section 4.15 of IEEE Std 279.

B. BRANCH TECHNICAL POSITION

1. If more restrictive safety trip points are required for operation with a reactor coolant pump out of service, and if operation with a reactor coolant pump out of service is of sufficient likelihod to be a planned mode of operation, the change to the more restrictive trip points should be accomplished automatically.

2. Plants with designs not in accordance with the above should have included in the plant technical specifications a requirement that the reactor be shut down prior to changing the set points manually.

C. REFERENCES

1. Millstone-3 Safety Evaluation Report, September 24, 1973.

2. Beaver Valley-2 Safety Evaluation Report, October 10, 1973.

3. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

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BRANCH TECHNICAL POSITION ICSB 13
DESIGN CRITERIA FOR AUXILIARY FEEDWATER SYSTEMS

A. <u>BACKGROUND</u>

The function of the auxiliary feedwater system in pressurized water reactors is to provide an emergency source of feedwater supply to the steam generators. It is required to ensure safe shutdown in the event of a main turbine trip with loss of offsite power. The system is also started on a safety injection signal. Feedwater is pumped to each steam generator through normally open control valves. It was found that in some plant designs the auxiliary feedwater system did not meet the single failure criterion. It is the purpose of this Branch Technical Position to provide guidance and to establish uniform requirements for acceptable designs of auxiliary feedwater systems.

B. <u>BRANCH TECHNICAL POSITION</u>

The auxiliary feedwater system should be capable of satisfying the system functional requirements after a postulated break in the auxiliary feedwater piping <u>inside</u> containment together with a single electrical failure. The basis for the position is that an auxiliary feedwater piping break would result in tripping the unit and, in turn, might cause loss of offsite power. Standard staff assumptions for analyzing postulated accidents include the assumption of loss of offsite power if the affected unit generator is tripped by the accident. Such a circumstance would leave the plant without adequate means for removal of afterheat even though the reactor coolant pressure boundary was intact, an unacceptable result. Plant heat removal systems must, in any postulated piping break, be capable of removing afterheat to the ultimate heat sink assuming a single electrical (active) failure anywhere in the auxiliary feedwater system or in the onsite power system.

C. <u>REFERENCES</u>

None

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BRANCH TECHNICAL POSITION ICSB 14

SPURIOUS WITHDRAWALS OF SINGLE CONTROL RODS IN PRESSURIZED WATER REACTORS

A. BACKGROUND

Recent operating experience with PWRs and subsequent reviews of PWR designs with regard to the requirements of General Design Criteria 20 and 25 have shown that single failures can cause inadvertent single rod withdrawals. The intent of this Branch Technical Position is to provide specific guidance toward an acceptable interpretation and application of GDC 20 and 25.

B. BRANCH TECHNICAL POSITION

Applicants have to demonstrate compliance with the requirements of GDC 20 to 25. For this purpose, it has to be shown by analysis that the consequences of uncontrolled or erruneous withdrawal of a single control rod under any possible conditions of reactor operation does not result in exceeding specified acceptable fuel design limits. If the results of this analysis show that the limits may be exceeded, the applicant must provide the results of failure modes and effects analyses to show that a single failure occurring in the control system, or an operator error, will not cause the uncontrolled or erroneous withdrawal of a single control rod. If the results of these analyses show that it is possible for uncontrolled or erroneous withdrawal of single control rods to occur, and the specified fuel design limits could be exceeded as a result, then the protection system must be designed to detect and terminate the resulting transient before the fuel design limits are exceeded.

C. REFERENCES

1. Surry 3 and 4 Safety Evaluation Report, March 26, 1974.

2. General Design Criteria, Appendix A, 10 CFR 50.

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BRANCH TECHNICAL POSITION ICSB 16

CONTROL ELEMENT ASSEMBLY (CEA) INTERLOCKS IN COMBUSTION ENGINEERING REACTORS

A. BACKGROUND

Certain control element assembly interlocks provided in Combustion Engineering designs have not been treated as safety-related. It has been determined by the staff that, unless it can be shown by analysis that these interlocks are not required to assure fuel integrity, they should be treated as required for safety.

B. BRANCH TECHNICAL POSITION

The following interlocks in CE designs are considered safety-related, and unless it can be substantiated otherwise by supporting analyses, they should be designed to meet the requirements of IEEE Std 279. The interlocks in question are intended to prevent the following actions:

1. Insertion of shutdown CEAs before the regulating CEAs are inserted.

2. Simultaneous withdrawal of more than two groups of CEAs.

3. Withdrawal of a CEA group or groups out of proper sequence.

C. REFERENCE

1. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

BRANCH TECHNICAL POSITION ICSB 19

ACCEPTABILITY OF DESIGN CRITERIA FOR HYDROGEN MIXING AND DRYWELL VACUUM RELIEF SYSTEMS

A. BACKGROUND

Certain design problems arise from the containment design concept which utilizes a drywell and suppression pool for heat removal after a loss-of-coolant accident (LOCA). Two such problems are (1) the hydrogen concentration in the drywell may, in a relatively short time, exceed the limits described in BTP CSB 6-2 (a safety-related problem), and (2) eventual cooling of the drywell will cause steam to condense, resulting in a partial vacuum which can draw water from the suppression pool and partially flood the drywell (a problem related to equipment deterioration and repair costs, not safety).

A hydrogen mixing system is proposed to mix the atmosphere in the larger containment volume outside the drywell with that in the drywell, thereby reducing the overall hydrogen concentration to an acceptable level. In some designs, the hydrogen mixing system bypasses the suppression pool, resulting in an additional load on the containment heat removal system, and in the possibility of overpressurizing the containment. (There are times during a LOCA when bypassing the suppression pool would quickly overpressurize the containment.)

Some designs propose to avoid flooding of the drywell by means of a vacuum relief system utilizing the valves of the hydrogen mixing system.

In view of the stresses to which the reactor operator might be subject during and following a LOCA, it has been concluded that automatic as well as manual initiation at the system level should be provided in BWR 6/Mark III plants.

B. BRANCH TECHNICAL POSITION

1. The design of the hydrogen mixing system should provide for both manual and automatic initiation and should conform to all criteria for protection systems, including the provisions of IEEE Std 279 and Regulatory Guides 1.22 and 1.62. Automatic initiation should come from the sensors which sense that the hydrogen concentration in the drywell has exceeded the limits described in BTP CSB 6-2.

2. The design should provide interlocks in both the automatic and manual circuits that will preclude the opening of valves which bypass the suppression pool before blowdown is complete.

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3. If the hydrogen mixing system bypasses the suppression pool, the containment heat removal system should be automatically initiated whenever the hydrogen mixing system is initiated.

4. The containment heat removal system should be automatically initiated upon indication of high pressure in the containment.

5. In conformance with paragraph 4.8 of IEEE Std 279, all signal inputs to the hydrogen mixing system and to those portions of the vacuum relief system which are common to the hydrogen mixing system, should be direct measures, to the extent practical, of the desired variable. Exceptions should be identified and justified.

C. <u>REFERENCES</u>

1. Branch Technical Position CSB 6-2, "Guidelines for the Evaluation of the Bypass Leakage in Dual Containment Plants," attached to Standard Review Plan 6.2.5.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. Regulatory Guide 1.62, "Manual Initiation of Protection Actions."

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BRANCH TECHNICAL POSITION ICSB 20
DESIGN OF INSTRUMENTATION AND CONTROLS PROVIDED TO ACCOMPLISH CHANGEOVER FROM INJECTION TO RECIRCULATION MODE

A. BACKGROUND

Designs are reviewed with regard to the automatic and manual initiation of protective actions, as set forth in paragrapn 4.17 of IEEE Std 279. For some recent designs, the staff concluded that the proposed design of the circuits used to change over to the recirculation mode of operation following a loss-of-coolant accident did not conform to IEEE Std 279, and the complexity of the proposed changeover procedure raised questions as to whether the operator could be expected to perform correctly the required actions within the time and based on the information available to him.

B. BRANCH TECHNICAL POSITION

1. A design that provides manual initiation at the system level of the transfer to the recirculation mode, while not ideal, is sufficient and satisfies the intent of IEEE Std 279 provided that adequate instrumentation and information display are available to the operator so that he can make the correct decision at the correct time. Furthermore, it should be shown that, in case of operator error, there are sufficient time and information available so that the operator can correct the error, and the consequences of such an error are acceptable.

2. Automatic transfer to the recirculation mode is preferable to manual transfer, for the reasons cited above, and should be provided for standard plant designs submitted for review on a generic basis under the Commission's standardization policy.

C. REFERENCES

1. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

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BRANCH TECHNICAL POSITION ICSB 21

GUIDANCE FOR APPLICATION OF REGULATORY GUIDE 1.47

A. BACKGROUND

The recommendations of Regulatory Guide 1.47 need further detailing as to methods of providing an acceptable design for the bypass and inoperable status indicators for engineered safety feature (ESF) systems. The purpose of this Branch Technical Position is to provide supplemental guidance for implementation of the recommendations of Regulatory Guide 1.47.

B. BRANCH TECHNICAL POSITION

The design criteria for bypass and inoperable status indication systems for ESF should reflect the importance of providing accurate information for the operator and reducing the possibility for the indicating equipment to affect adversely the monitored safety systems. In developing the design criteria, the following should be considered:

1. The bypass indicators should be arranged to enable the operator to determine the status of each safety system and determine whether continued reactor operation is permissible.

2. When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.

3. Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design.

4. Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety. Administrative procedures should not require immediate operator action based soley on the bypass indications.

5. The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on plant safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems.

6. The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified.

C. REFERENCES

1. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

BRANCH TECHNICAL POSITION ICSB 22
GUIDANCE FOR APPLICATION OF REGULATORY GUIDE 1.22

A. BACKGROUND

A recent application listed eight functions that are not tested while the reactor is operating at power. The applicant claimed that the periodic testing complied with Regulatory Guide 1.22. Regulatory Guide 1.22 does make provisions for actuated equipment that is not tested during reactor operation but it does not have provisions for excluding any portion of the protection system from the requirements of paragraphs 4.9 and 4.10 of IEEE Std 279.

B. BRANCH TECHNICAL POSITION

All portions of the protection systems should be designed in accordance with IEEE Std 279, as required by 10 CFR §50.55a(h). All actuated equipment that is not tested during reactor operation should be identified and a discussion of how each conforms to the provisions of paragraph D.4 of Regulatory Guide 1.22 should be submitted.

C. REFERENCES

1. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

2. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

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BRANCH TECHNICAL POSITION ICSB 25

GUIDANCE FOR THE INTERPRETATION OF GENERAL DESIGN CRITERION 37 FOR TESTING THE OPERABILITY OF THE EMERGENCY CORE COOLING SYSTEM AS A WHOLE

A. BACKGROUND

General Design Criterion 37 requires, in part, that the emergency core cooling system be designed to permit testing the operability of the system as a whole under conditions as close to design as practical. It is stated in one recent application that the safety injection and residual heat removal pumps are made inoperable during the system tests.

B. BRANCH TECHNICAL POSITION

In order to comply with the requirements of GDC 37, all ECCS pumps should be included in the system test.

C. REFERENCES

1. General Design Criteria, Appendix A, 10 CFR 50.

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BRANCH TECHNICAL POSITION ICSB 26
REQUIREMENTS FOR REACTOR PROTECTION SYSTEM ANTICIPATORY TRIPS

A. BACKGROUND

Several reactor designs have incorporated a number of anticipatory or "back-up" trips for which no credit was taken in the accident analyses. These trips, as a rule, were not designed to the requirements of IEEE Std 279 and therefore introduced nonsafety grade equipment into the reactor protection system. It was determined by the staff that this was not an acceptable practice, because of possible degradation of the reactor protection system.

B. BRANCH TECHNICAL POSITION

All reactor trips incorporated in the reactor protection system should be designed to meet the requirements of IEEE Std 279, without exception. This position applies to the entire trip function from the sensor to the final actuated device.

C. REFERENCES

1. Shearon Harris Safety Evaluation Report, September 15, 1972.

2. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

APPENDIX 7-B GENERAL AGENDA, STATION SITE VISITS

An important part of the review at the operating license stage is a site visit. It is preferable to have the site visit sometime before the completion of the drawing review. The purpose of the site visit is to supplement the review of the design based on the drawings and to evaluate the actual implementation of the design as installed at the site. The Regional Office of Regulatory Operations having jurisdiction over the plant under consideration should be notified ahead of time of the visit so that the regional inspectors can become familiar on a first-hand basis with findings that may require followup action. Since proper implementation of design is the ultimate goal of the technical review process, the importance of a site visit is self-evident. The following is a typical general agenda that may be used as a guide for developing a specific agenda for the plant under review.

1. <u>Preliminary Discussions</u>
   a. Unresolved items.
   b. Plant layout for touring.
   c. Special interest areas.

2. <u>Control Room</u>
   a. General layout.
   b. Nuclear and reactor protection instrument arrangement and layout.
   c. Rod position indication.
   d. Protection system initiation and bypass switch arrangements.
   e. Diesel control board.
   f. Cabling in control room (separation, loading, etc.).
   g. Radiation monitoring.
   h. Engineered safety feature initiation and bypass switch arrangements and status panels.

3. <u>Cable Runs and Cable Spreading Area</u>
   a. General layout.
   b. Degree of separation.
   c. Diverse wiring.
   d. Tray or wireway density (percentage fill).

e. Fire detection and protection.

f. Penetrations and cable terminations.

4. Switchgear Rooms

a. General layout.

b. Physical and electrical separation of redundant units.

c. Potential for damage due to fire, missiles, etc.

d. Cable installation.

e. Fire detection and protection.

5. Battery Installations

a. General layout.

b. Physical and electrical separation.

c. Potential for damage due to fire, missiles, etc.

d. Fire detection and protection and security.

e. Ventilation independence.

f. Monitoring instrumentation.

6. Diesel Generators

a. General layout.

b. Physical and electrical separation of redundant units.

c. Fuel supply system.

d. Fire detection and protection.

e. Qualification tests - interlocks and control panel.

f. Auxiliary systems - starting air, combustion air, ventilation.

7. Instrument Piping

a. Physical separation and single failure.

b. Potential for damage due to fire, flooding, etc.

c. Test features.

8. Transformers (Switchyard)

a. Physical and electrical separation.

b. Potential for damage due to fire, flooding, missiles, etc.

c. Fire detection and protection.

9. Quality Control

a. Onsite receipt, storage, installation, and protection procedures of installed instrumentation, equipment, and cables.

10. Reactor Building and Turbine Building

a. Protection system instrument arrangement and layout.

b. Potential for instrument damage due to fire, missiles, etc.

c. Separation of piping and wiring to redundant instruments.

d. Provisions for testing protection instruments.

11. Shared Systems for Multi-Unit Sites
    a. Equipment location and potential for damage.
    b. Control room control and assignment to accident unit.
    c. Availability upon completion of first unit.

12. Steam Lines - Main, HPCI, RCIC
    a. BWR temperature and radiation monitoring systems.
    b. Isolation valves.

13. Recirculation Water System (Condenser)
    a. Break detection and flood protection features.

14. Shutdown Outside Control Room
    a. Location for potential damage.
    b. Feedwater system, etc.

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