U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

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 the guidelines provided in regulatory guides and 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 is identified parenthetically as part of the BTP designation.

BTP ICSB	Branch Technical Positions of the Instrumentation and Control Systems Branch
1.	Deleted
3.	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System.
4.	Requirements of Motor-Operated Valves in the ECCS Accumulator Lines.
5.	[BTP has been superseded by Standard Technical Specifications.]

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9. [BTP has been superseded by Standard Technical Specifications.]

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.

16. Deleted.

19. Deleted.

20. Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode.

21. Guidance for Application of Regulatory Guide 1.47.

22. Guidance for Application of Regulatory Guide 1.22.

25. [BTP has been superseded by Standard Technical Specifications.]

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

[BTP ICSB 1 has been deleted]

BRANCH TECHNICAL POSITION ICSB 3
ISOLATION OF LOW PRESSURE SYSTEMS FROM THE HIGH PRESSURE REACTOR COOLANT SYSTEM

A. BACKGROUND

During normal 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. The residual heat removal system used for cold shutdown conditions when in service becomes an extension of the reactor coolant pressure boundary. General Design Criteria 15 requires that reactor coolant system and associated auxiliary, control and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. 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.

6. The system should satisfy the requirements of the General Design Criteria and Section 50.55a(h) of 10 CFR Part 50 with regards to the protection system requirements (IEEE Std 279).

C. REFERENCES

None

BRANCH TECHNICAL POSITION ICSB 4
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. Bypassed and inoperable status indication in accordance to Regulatory Guide 1.47.

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).

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

[BTP has been superseded by Standard Technical Specifications]

BRANCH TECHNICAL POSITION ICSB 9
DEFINITION AND USE OF "CHANNEL CALIBRATION" - TECHNICAL SPECIFICATIONS

[BTP has been superseded by Standard Technical Specifications]

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 likelihood 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."

BRANCH TECHNICAL POSITION ICSB 13
DESIGN CRITERIA FOR AUXILIARY FEEDWATER SYSTEMS

A. BACKGROUND

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. BRANCH TECHNICAL POSITION

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. REFERENCES

None

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

GDC 20 requires that the protection system shall be designed to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences. GDC 25 requires these limits shall not be exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection) of control rods. Within the context of GDC 20 the staff considers operator error to be an anticipated operational occurrence, in addition to the consideration of single malfunction requirements of GDC 25, for which conformance to these requirements is to be evaluated. The applicant should perform analyses of the reactivity control systems* and analyze the consequences of operator error to assess the impact of these events on fuel design limits. If the results of these analyses show that specified acceptable fuel design limits may be exceeded for these events, the protection system must be designed to detect and terminate these events prior to exceeding these limits.

With regards to the evaluation of malfunctions within the reactivity control systems, consideration should be given to failures which cause actions as well as prevent actions such that all possible effects are examined. Further, consideration of failures which could lead to single or multiple rod position changes or out of sequence rod patterns should be analyzed, as well as, failures which could lead to reactivity changes by boron control systems.

C. REFERENCES

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

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

*Reactivity control systems include interlocks within the system which limit the consequences of control system failures.

BRANCH TECHNICAL POSITION ICSB 16
CONTROL ELEMENT ASSEMBLY (CEA) INTERLOCKS IN COMBUSTION ENGINEERING REACTORS

[BTP has been deleted]

BRANCH TECHNICAL POSITION ICSB 19
ACCEPTABILITY OF DESIGN CRITERIA FOR HYDROGEN MIXING AND DRYWELL VACUUM RELIEF SYSTEMS

[BTP has been deleted]

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 paragraph 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."

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 solely 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 Part 50, §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."

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

[BTP has been superseded by Standard Technical Specifications]

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. This position applies to the entire trip function from the sensor to the final actuated device. For sensors located in nonseismic areas the installation (including circuit routing) and design should be such that the effects of credible faults (i.e. grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the RPS and degrade the RPS performance or reliability. The sensors should be qualified to operate in a seismic event, i.e., not fail to initiate a trip for conditions which would cause a trip.

C. REFERENCES

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

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

3. NUREG 0718 and 0737, Items II.K.2.10* and II.K.3.12.

*Task Action Plan item II.K.1.21 has been superseded by item II.K.2.10.