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

SECTION 7.1 INTRODUCTION

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

Section 7.1 of the applicant's safety analysis report (SAR) contains information pertaining to safety-related instrumentation and control systems, their design bases, and the applicable acceptance criteria. EICSB reviews this information as detailed in III of this plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format (Item 4.1 of Ref. 1).

The secondary review branches (APCSB, CSB, RSB) review the safety-related system tabulations for completeness, i.e., to verify that all safety-related systems within their respective areas of primary review responsibility have been identified. If systems other than those identified are deemed to be safety-related, this information is transmitted to EICSB.

This review plan also includes evaluation of the proposed technical specifications given in SAR Chapter 16 to assure that they are adequate with regard to safety system settings, limiting conditions for operation, and periodic surveillance testing of instrumentation and controls.

II. ACCEPTANCE CRITERIA

The identification of safety-related systems is acceptable when it can be concluded that the integrated response of these systems assures the safety of the plant in normal operation, anticipated operational transients, and postulated accidents.

Table 7-1, "Acceptance Criteria for Controls," lists the criteria currently applicable to safety-related instrumentation and control systems (acceptance criteria for safety-related electric power systems are listed in Table 8-1). Conformance to these criteria does not necessarily establish the adequacy of the functional performance and reliability of these systems. However, omission of any of the criteria will in most cases be an indication of system inadequacy. Therefore, the identification of the criteria applicable to safety-related instrumentation and control systems is acceptable if it includes all of the criteria listed in Table 7-1, and if the SAR contains a statement to the effect that these criteria are implemented, at the operating license (OL) stage, or will be implemented, at the construction permit (CP) stage, in the design of these systems.

The fundamental bases for acceptance of the proposed technical specifications are that the limiting conditions for operation are such that sufficient equipment is required to be available for operation to meet the single failure criterion; that equipment outages that are permissible for a short period of time still leave available sufficient equipment to provide the protective function assuming no failures; and that the provisions of the technical specifications are compatible with the safety analyses.

III. REVIEW PROCEDURES

Safety-related systems fall into three categories: basic safety systems, auxiliary supporting systems, and other systems important to safety.

Basic safety systems are those that directly perform a protective function. Examples are the reactor trip system, the emergency core cooling system, the containment isolation system, and the containment spray system. The reactor trip system provides reactor protection by fast insertion of negative reactivity (control rods) when plant conditions approach design safety limits. All the other systems listed are engineered safety features (ESF) systems; their function is to mitigate the consequences of postulated design basis accidents.

Auxiliary supporting systems are those that must function to enable operation of the basic safety systems. Component cooling systems, service water systems, ventilation systems, and electric power systems which serve ESF and reactor trip components are examples of auxiliary supporting systems. These systems must meet the same criteria as the basic safety systems they support.

Other systems important to safety are those systems which operate to reduce the probability of occurrence of specific accidents, or to maintain the plant (including other safety systems) within the envelope of operating conditions postulated in the accident analyses as being required to assure full protection capability. Examples of this type of system are the cold loop startup control (interlocks) system, the accumulator tank isolation valve control (interlocks, position indication, alarms) system, and the plant status and alarm systems that provide the operator with the information necessary for initiating manual protective action. These systems are primarily instrumentation and control systems characterized by having a functional interface with the operator. The same safety criteria apply. However, in application to this type of system, the criteria are usually further defined by regulatory guides and in branch technical positions of the EICSB.

The EICSB review encompasses all of the electric power, instrumentation, and control systems associated with all three categories of safety-related systems described above, with particular emphasis on the elements which constitute the protection system (as defined in IEEE Std 279-1971) and the Class IE electric systems (as defined in IEEE Std 308-1971). The safety-related electric power systems are covered in the standard review plans for Chapter 8 of the SAR. The standard review plans for SAR Chapter 7 are concerned only with the safety-related instrumentation and control systems.

The review of SAR Section 7.1 and applicable portions of the plant technical specifications is performed as follows:

1. EICSB will establish that all safety-related systems are identified, and that this identification does not conflict with the more detailed information provided in other sections of the SAR, particularly in Chapters 6 and 8 and in subsequent sections of Chapter 7. The definitions of safety-related systems presented above should be used as an aid in assessing the completeness of the identification. The secondary review branches (APCSB, CSB, RSB) will confirm the identification of all safety-related systems within their respective areas of primary review responsibility. If systems other than those identified are deemed to be safety-related, this information should be transmitted to EICSB. Particular care should be exercised to assure that all systems postulated in the accident analyses (Chapter 15) as being required for safety are identified as safety-related systems.

2. EICSB verifies that other systems described in the SAR (particularly in Chapters 5, 6, 8, 9, 10, 11, and 15) but not identified by the applicant in Section 7.1 are not required for safety. The reviewer should obtain concurrence from the secondary review branches with regard to systems considered to be safety-related by EICSB, but which have not been identified as such by the applicant. Written requests for evaluation should be made to the secondary review branches when there are novel designs or significant differences of opinion.

3. EICSB verifies that the safety-related systems are categorized by supplier, i.e., those designed and supplied by the nuclear steam system supplier and those designed or supplied by others.

4. EICSB verifies that systems identical to those of reference plants that have recently received construction permits or operating licenses and those that differ from the reference plants are so identified.

5. EICSB verifies that for those systems that are different from those of the reference plants, the differences are described and justified to the extent necessary for an evaluation of their safety significance.

6. EICSB confirms that the criteria identified as being applicable to the design of safety-related instrumentation and control systems include those criteria listed in Table 7-1. This identification meets the applicable requirements of General

Design Criterion 1, "Quality Standards and Records," of Appendix A of 10 CFR Part 50. General Design Criterion 1 also requires that, "Structures, systems and components important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed." Therefore, the SAR should include (1) a discussion regarding the applicability of each criterion listed, and (2) a statement to the effect that the criteria are implemented (OL) or will be implemented (CP) in the design of safety-related instrumentation and control systems.

7. EICSB verifies that technical design bases are provided (reference to other sections of the SAR is acceptable) for all the various functions of the protection system.

8. Applicable portions of the proposed plant technical specifications (SAR Chapter 16) are reviewed by EICSB and the secondary review branches to:
   a. Confirm the suitability of the safety limits, limiting safety system settings, and the limiting conditions for operation.
   b. Verify that the frequency and scope of periodic surveillance requirements are adequate.

For a CP review, it is only necessary to confirm that the applicant has identified those variables, conditions, or other items which have been determined to be probable subjects of the technical specifications (See 10 CFR §50.34(a)(5).). The applicant's justification for the selection of those items is evaluated with special attention to any that may significantly influence the final design. The specific provisions of the proposed technical specifications are not approved during the CP review. However, any specific provisions which are known to be unacceptable or which may influence acceptance of the preliminary design of the plant should be brought to the applicant's attention and, if approrpiate, included in that portion of the staff's safety evaluation report pertaining to the design of the affected systems.

For an operating license review, the proposed technical specifications are reviewed and evaluated in depth in accordance with the requirements of 10 CFR §50.36. For the EICSB areas of review, a check is made that the limiting conditions for operation (LCO) agree with the surveillance requirements, i.e., for each system or component that is the subject of a LCO, there must be corresponding surveillance requirements. Each system or component that performs a function for which credit is taken in the accident analyses should be the subject of an LCO. The limiting safety system settings should be in accordance with the values assumed in the accident analyses, including appropriate allowances for instrument error, drift, etc. If the acceptance of the design of a particular system is based upon required plant conditions or particular operating procedures, such requirements should be included in the final technical specifications and, if appropriate, noted in that portion of the staff's safety evaluation report pertaining to the design of the affected system.

IV. EVALUATION FINDINGS

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

> "The applicant has identified the safety-related instrumentation and control systems and the applicable safety criteria and has documented his intent to design and implement these systems in accordance with the criteria. It is concluded that implementation of these systems in accordance with the criteria provides assurance that the plant will perform as designed in normal operation, anticipated operational transits, and postulated accident conditions, and meets the applicable requirements of General Design Criterion 1."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."*

*All references for this plan are included in Standard Review Plan Table 7-1.

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SRP Table 7-1

SRP Table 7-1

U.S. NUCLEAR REGULATORY COMMISSION

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TABLE 7-1

ACCEPTANCE CRITERIA FOR CONTROLS

Table 7-1 contains the acceptance criteria for the review plans of Chapter 7. These acceptance criteria include the applicable general design criteria, IEEE standards, regulatory guides, and branch technical positions (BTP) of the Electrical, Instrumentation and Control Systems Branch (EICSB). The table was prepared by EICSB for use by its members in reviewing Chapter 7 and for use by secondary review branch reviewers.

The applicability of these criteria to specific sections of Chapter 7 is indicated by an X in the matrix listing of criteria and SAR sections. There is a corresponding table (8-1) at the end of Chapter 8 covering the acceptance criteria of safety-related power supplies. The BTP listed in Tables 7-1 and 8-1 are contained in Appendix 7-A to the Chapter 7 review plans.

ACCEPTANCE CRITERIA FOR CONTROLS - TABL. 7-1

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
1. 10 CFR Part 50									
a. 10 CFR §50.34	Contents of Application: Technical Information	X	X	X	X	X	X	X	
b. 10 CFR §50.36	Technical Specifications	X	X	X	X	X	X		
c. 10 CFR §50.55a	Codes and Standards	X	X	X	X	X	X	X	
2. General Design Criteria (GDC), Appendix A to 10 CFR Part 50									
a. GDC 1	Quality Standards and Records	X	X	X	X	X	X		
b. GDC 2	Design Bases for Protection Against Natural Phenomena	X	X	X	X	X	X		
c. GDC 3	Fire Protection	X	X	X	X	X	X		
d. GDC 4	Environmental and Missile Design Bases	X	X	X	X	X	X		
e. GDC 5	Sharing of Structures, Systems, and Components	X	X	X	X	X	X		
f. GDC 10	Reactor Design	X	X	X	X	X	X		
g. GDC 12	Suppression of Reactor Power Oscillations	X	X			X		X	
h. GDC 13	Instrumentation and Control	X	X	X	X	X	X	X	
i. GDC 15	Reactor Coolant System Design	X	X			X		X	
j. GDC 19	Control Room	X	X	X	X	X	X	X	
k. GDC 20	Protection System Functions	X	X	X	X	X	X		
l. GDC 21	Protection System Reliability and Testability	X	X	X	X	X	X		
m. GDC 22	Protection System Independence	X	X	X	X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
n. GDC 23	Protection System Failure Modes	X	X	X	X	X	X		
o. GDC 24	Separation of Protection and Control Systems	X	X	X	X	X	X	X	
p. GDC 25	Protection System Requirements for Reactivity Control Malfunctions	X	X			X			
q. GDC 26	Reactivity Control System Redundancy and Capability	X	X		X	X		X	
r. GDC 27	Combined Reactivity Control Systems Capability	X	X		X	X		X	
s. GDC 28	Reactivity Limits	X	X			X	X	X	
t. GDC 29	Protection Against Anticipated Operational Occurrences	X	X	X	X	X	X	X	
u. GDC 33	Reactor Coolant Makeup	X			X	X	X		
v. GDC 34	Residual Heat Removal	X		X	X	X	X		
w. GDC 35	Emergency Core Cooling	X	X	X		X	X		
x. GDC 37	Testing of Emergency Core Cooling System	X	X	X		X	X		
y. GDC 38	Containment Heat Removal	X		X		X	X		
z. GDC 40	Testing of Containment Heat Removal System	X		X		X	X		
aa. GDC 41	Containment Atmosphere Cleanup	X		X		X	X		
bb. GDC 43	Testing of Containment Atmosphere Cleanup Systems	X		X		X	X		
cc. GDC 44	Cooling Water	X		X		X	X		
dd. GDC 46	Testing of Cooling Water System	X		X		X	X		
ee. GDC 50	Containment Design Basis	X				X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
ff. GDC 54	Piping Systems Penetrating Containment	X		X		X	X		
gg. GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	X		X		X	X		
hh. GDC 56	Primary Containment Isolation	X		X		X	X		
ii. GDC 57	Closed Systems Isolation Valves	X		X		X	X		
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:									
a. IEEE Std 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See 10 CFR §50.55a(h) and Reg. Guide 1.62
b. IEEE Std 308-1971	Criteria for Class IE Electric Systems for Nuclear Power Generating Stations	X			X	X	X		See Reg. Guide 1.32.
c. IEEE Std 317-1972	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.63.
d. IEEE Std 336-1971 (ANSI N45.2.4-1972)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.30.
e. IEEE Std 338-1971	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X		
f. IEEE Std 344-1971 (ANSI N41.7)	Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations	X	X	X	X	X	X		
g. IEEE Std 379-1972 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Sytems	X	X	X	X	X	X	X	See Reg. Guide 1.53.
h. IEEE Std 384-1974 (ANSI N41.14)	Criteria for Separation of Class IE Equipment and Circuits	X	X	X		X			

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
		APPLICABILITY (SAR Section)							
4. Regulatory Guides (RG)									
a. RG 1.6	Independence Between Reduncant Standby (Onsite) Power Sources and Between Their Distribution Systems	X			X		X		
b. RG 1.7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident	X		X		X			
c. RG 1.11	Instrument Lines Penetrating Primary Reactor Containment	X	X	X	X	X	X		
d. RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	X	X		
e. RG 1.29	Seismic Design Classification	X	X	X	X	X	X		
f. RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	X	X	X	
g. RG 1.32	Use of IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"	X			X	X	X		
h. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	X	X		Use in conjunction with Position 3, RG 1.17
i. RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X	X	X	X	X	X		
j. RG 1.62	Manual Initiation of Protection Actions	X	X	X	X		X		
k. RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	X	X	X	X	X	X	X	
l. RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X						

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
m. RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2.	X	X	X	X	X	X	X	
n. RG 1.75	Physical Independence of Electric Systems	X	X	X		X			
o. RG 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	X					X		
p. RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X	X	X	X	X	X		
q. RG 1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	X		X					
5. Branch Technical Positions (BTP) EICSB									
a. BTP EICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X	X	X	X		X		
b. BTP EICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System	X			X		X		
c. BTP EICSB 4	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	X			X		X		
d. BTP EICSB 5	Scram Breaker Test Requirements - Technical Specifications	X	X						
e. BTP EICSB 9	Definition and Use of "Channel-Calibration" - Technical Specifications	X	X		X	X	X		
f. BTP EICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X	X		X	X	X		
g. BTP EICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	X	X	X					

TABLE 7-1 (CONTINUED)

	CRITERIA	TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
h.	BTP EICSB 13	Design Criteria for Auxiliary Feedwater Systems	X		X					
i.	BTP EICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	X	X						
j.	BTP EICSB 15	Reactor Coolant Pump Breaker Qualification	X	X						
k.	BTP EICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors	X	X						
l.	BTP EICSB 18	Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves	X		X	X		X		
m.	BTP EICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems	X		X			X		
n.	BTP EICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recriculation Mode	X		X	X		X		
o.	BTP EICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X	X	X		
p.	BTP EICSB 22	Guidance for Application of Reg. Guide 1.22	X	X	X	X	X	X		
q.	BTP EICSB 23	Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown	X				X			
r.	BTP EICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times	X	X	X	X		X		
s.	BTP EICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole	X		X	X				
t.	BTP EICSB 26	Requirements for Reactor Protection System Anticipatory Trips	X	X						
u.	BTP EICSB 27	Design Criteria for Thermal Overland Protection for Motors of Motor-Operated Valves	X		X	X		X		

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SRP 7.2