Table A.0-1

AEC GENERAL DESIGN CRITERIA - GROUP I
(OVERALL PLANT REQUIREMENTS)

| Criterion | | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 1. | Quality Standards | 1.5, 1.10, 2.0, 3.2, 3.3, 3.4, 3.5, 3.7, 3.8, 4.2, 4.3, 4.4, 4.5, 4.6, 4.7, 4.8, 5.0, 6.0, 7.2, 7.3, 8.0, App. D | See Note 1 |
| 2. | Performance Standards | 1.5, App C. | See Note 2 |
| 3. | Fire Protection | 5.0, 10.11, 12.0 | |
| 4. | Sharing of Systems | 2.6, 5.3, 7.14, 7.15, 7.16, 8.4, 8.5, 8.6, 9.2, 9.3, 9.4, 10.5, 10.6, 10.7, 10.8, 10.9, 10.10, 10.11, 10.12, 10.14, 11.6, 12.2, App F., 4.8 | See Note 3 |
| 5. | Records Requirements | 13.7, App D | |

Notes: 1. The phrase in the criterion statement "are essential to" is interpreted as "have a vital role in." The last sentence is interpreted to mean that where such programs and procedures are not covered by applicable codes and standards, a showing of sufficiency is required.

2. As in Criterion 1 phase "have a vital role in" is interpreted from "are essential to" in the first sentence. The latter part of the sentence is interpreted as "to withstand, without undue risk to the health and safety of the public, the forces that might be imposed by the occurrence of natural phenomena such as earthquakes, tornadoes, flooding conditions, wind or ice."

3. The following rewording emphasizes that this criterion applies only to critical safety systems and recognizes that the intent is not to restrict all sharing of systems but only to restrict sharing where there would be a significant degradation of safety: "Where reactor facilities share systems or components which have a critical safety function, it shall be shown that safety is not significantly impaired by the sharing."

Table A.0-2

AEC GENERAL DESIGN CRITERIA - GROUP II (PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS)

| Criterion | | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 6. | Reactor Core Design | 1.5, 1.7, 3.2, 3.6, 3.7, 4.3, 4.7, 4.8, 7.2, 14.2, 14.4, 14.5, 14.6 | |
| 7. | Suppression of Power Oscillations | 1.5, 3.4, 3.6, 3.7, 4.4, 7.2, 7.5, 7.7, 7.17, 14.5 | |
| 8. | Overall Power Coefficient | 1.5, 1.7, 3.6, 3.7, 7.17 | |
| 9. | Reactor Coolant Pressure Boundary | 1.5, 4.2, 4.3, 4.4, 4.10, 4.11, 7.8, 14.5, 14.6, App.C | |
| 10. | Containment | 5.2, 5.3, 14.4, 14.6 | |

Table A.0-3

AEC GENERAL DESIGN CRITERIA - GROUP III
(NUCLEAR AND RADIATION CONTROLS)

| | Criterion | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 11. | Control Room | 1.5, 7.2-7.5, 7.7-7.10, 7.12, 12.2, 12.3 | See Note 1 |
| 12. | Instrumentation and Control System | 1.5, 3.4, 3.8, 4.10, 7.2, 7.3, 7.4, 7.5, 7.7, 7.8, 7.9, 7.10, 7.12, 7.13, 7.14, 7.16, 9.2, 9.3, 9.4 | See Note 2 |
| 13. | Fission Process Monitors and Controls | 1.5, 3.4, 3.8, 7.2, 7.5, 7.7, 7.8, 7.9, 7.16 | |
| 14. | Core Protection System | 1.5, 3.4, 3.5, 4.4, 4.5, 4.6, 4.7, 4.8, 6.1-6.7, 7.2-7.5, 7.7, 7.12, 14.1-14.7 | |
| 15. | Engineered Safety Features Protection Systems | 1.5, 7.2-7.5, 7.12 | |
| 16. | Monitoring Reactor Coolant Pressure Boundary | 1.5, 4.10, 5.2, 7.8, 10.16 | See Note 3 |
| 17. | Monitoring Radioactive Releases | 1.5, 7.12, 7.13, 7.14, 9.2, 9.4, | |
| 18. | Monitoring Fuel and Storage | 7.6, 7.12, 7.13, 9.2, 9.4, 10.5 | See Note 4 |

Notes:
1. It is assumed that the event which renders the control room unhabitable does not occur simultaneously with a reactor accident.
2. Although the variables referred to in the criterion are not specified, the intent is taken to mean only those variables are included which are associated with reactor control, are measurable, could result in an unsafe condition, and have established operating limits.
3. This criterion is interpreted to mean that the coolant pressure boundary is monitored by the detection of leakage through it.
4. The words "contribute to" are interpreted as "result in" to improve clarity.

Table A.0-4

AEC GENERAL DESIGN CRITERIA - GROUP IV (RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS)

| Criterion | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|
| 19. Protection System Reliability | 1.5, 3.4, 7.2-7.5, 7.12, 14.0 | |
| 20. Protection Systems Redundancy and Independence | 1.5, 3.4, 7.2-7.5, 7.12, 14.0 | |
| 21. Single Failure Definition | 1.2, 14.4 | |
| 22. Separation of Protection and Control Instrumen-tation System | 1.5, 3.4, 7.2-7.5, 7.12 | |
| 23. Protection Against Multiple Disability for Protection Systems | 1.5, 3.4, 7.2-7.5, 7.12, 14.0 | |
| 24. Emergency Power for Protection Systems | 1.5, 3.4, 6.4, 7.2-7.5, 7.12, 8.4, 8.5, 14.0 | |
| 25. Demonstration of Functional | 1.5, 3.4, 4.6, 4.8, 5.2, 5.3, 6.7, 7.2-7.5, 7.12 | |
| 26. Protection Systems Fail-Safe Design | 1.5, 6.1-6.5, 7.2-7.5, 8.4, 8.5 | |

Table A.0-5

AEC GENERAL DESIGN CRITERIA - GROUP V (REACTIVITY CONTROL)

| Criterion | | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 27. | Redundancy of Reactivity Control | 1.5, 3.4, 3.8, 7.7 | |
| 28. | Reactivity Hot Shutdown Capacity | 1.5, 3.4, 3.6, 3.8, 7.7, 14.0 | See Note 1 |
| 29. | Reactivity Shutdown Capacity | 1.5, 3.4, 3.6, 7.2, 14.0 | |
| 30. | Reactivity Holddown Capacity | 1.5, 3.4, 3.6, 3.8 | |
| 31. | Reactivity Control Systems Malfunction | 1.5, 3.4, 3.6, 3.7, 7.2, 7.7, 14.0 | See Note 2 |
| 32. | Maximum Reactivity Worth of Control Rods | 1.5, 3.4, 3.6, 3.7, 7.7, 14.0 | See Note 3 |

Notes:

1. As presently worded, this criterion can be read to require that of fast scram. This is not consistent with current practice and is not taken to be the intent. It also fails to recognize the degree of reliability with which the primary reactivity shutdown system operates. By this interpretation that part of the sentence which follows the words "hot operating condition" is deleted.

2. The phrase in the criterion statement "in conjunction with the reactor protection systems" is inserted after the words, "reactivity control systems" in the first sentence to recognize that the reactor protection system is required to protect against certain control system malfunctions and operator errors. The parenthetical expression is expanded to "(not ejection or dropout)".

3. Item (b) in the criterion statement is interpreted to read "(b) disrupt the core, its support structures, or other vessel internals sufficiently to significantly impair the effectiveness of core cooling."

Table A.0-6

AEC GENERAL DESIGN CRITERIA - GROUP VI (REACTOR COOLANT PRESSURE BOUNDARY)

| Criterion | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|
| 33. Reactor Coolant Pressures Boundary Capability | 1.5, 3.3-3.6, 4.2, 4.4, 4.5, 4.6, 4.11, 14.4-14.6, App. C | |
| 34. Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention | 3.3, 4.2, 4.3, 7.8, App. C, App. D | See Note 1 |
| 35. Reactor Coolant Pressure Boundary Brittle Fractures Prevention | 4.2 | |
| 36. Reactor Coolant Pressure Boundary Surveillance | 4.2, 4.3, 4.10, 4.12 | |

Notes: 1. The interpretation of this criterion is that it should relate directly to codes and standards currently in existence or planned for the future. Accordingly, it is interpreted to read as follows: "The reactor coolant pressure boundary shall be designed and operated to reduce to acceptable levels the probability of rapidly propagating type failures. Consideration shall be given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes."

Table A.0-7

(Sheet 1)

AEC GENERAL DESIGN CRITERIA - GROUP VII (ENGINEERED SAFETY FEATURES)

| Criterion | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|
| 37. Engineered Safety Features Basis for Design | 1.5, 3.3, 3.4, 4.2, 4.4, 4.6, 5.2, 5.3, 6.1-6.7, 7.2, 7.3, 7.4, 8.4, 8.5, 8.6, 14.1-14.7 | See Note 1 |
| 38. Reliability and Testability of Engineered Safety Features | 1.5, 3.4, 3.5, 4.6, 5.2, 5.3, 6.6, 7.2, 7.3, 7.4, 7.5, 7.12, 8.4, 8.5, 8.6 | |
| 39. Emergency Power for Engineered Safety Features | 7.2, 7.3, 7.4, 8.5, 8.6 | See Note 2 |
| 40. Missile Protection | 5.2, 12.2, App.C | |
| 41. Engineered Safety Features Performance Capability | 6.1-6.5, 7.4, 14.1-14.6 | See Notes 2 and 3 |
| 42. Engineered Safety Features Components Capability | 3.4, 5.2, 5.3, 6.1-6.5, 7.2, 7.3, 7.4, 8.4, 8.5, 8.6, 14.6 | See Note 4 |
| 43. Accident Aggravation Protection | 3.4, 5.2, 5.3, 6.1-6.5, 7.3, 7.4, 8.4, 8.5, 8.6 | |
| 44. Emergency Core Cooling Systems Capability | 6.1-6.5, 7.4, 14.6 | See Note 5 |
| 45. Inspection of Emergency Core Cooling Systems | 3.3, 4.2, 4.12, 6.6 | |
| 46. Testing of Emergency Core Cooling Systems Components | 1.5, 6.6, 7.4 | |
| 47. Testing of Emergency Core Cooling Systems | 7.4, 6.6 | |
| 48. Testing of Operational Sequence of Emergency Core Cooling Systems | 1.5, 6.4, 6.6, 7.4, 8.5, 8.6, 10.9, 10.10 | See Note 6 |
| 49. Containment Design Basis | 1.5, 4.6, 5.2, 5.3, 6.1, 6.2, 6.5, 7.3, 7.4, 10.6, 10.9, 14.2, 14.7, App.C | See Note 7 |
| 50. NDT Requirement for Containment Material | 5.2 | See Note 8 |
| 51. Reactor Coolant Pressure Boundary Outside Containment | 1.5, 2.2, 2.3, 5.2, 4.6, 7.3, 14.6 | |
| 52. Containment Heat Removal Systems | 1.5, 4.8, 5.2, 6.1-6.5, 7.4, 10.6, 10.9, 14.6, 14.7 | |
| 53. Containment Isolation Valves | 1.5, 4.6, 5.2, 7.3 | |
| 54. Containment Leakage Rate Testing | 5.2 | |
| 55. Containment Periodic Leakage Rate Testing | 5.2 | See Note 9 |

(Sheet 2)

AEC GENERAL DESIGN CRITERIA - GROUP VII (ENGINEERED SAFETY FEATURES)

| Criterion | | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 56. | Provisions for Testing of Penetrations | 5.2, 5.3 | |
| 57. | Provisions for Testing of Isolation Valves | 4.6, 5.2, 7.3, 7.12 | |
| 58. | Inspection of Containment Pressure-Reducing Systems | 4.8, 5.2, 5.3, 6.4, 6.6, 10.6, 10.9, 12.2 | |
| 59. | Testing of Containment Pressure-Reducing Systems Components | 4.8, 5.2, 5.3, 6.4, 6.6, 7.3, 7.4, 10.6, 10.9 | |
| 60. | Testing of Containment Spray Systems | 6.4, 6.6, 7.4 | |
| 61. | Testing of Operational Sequence of Containment Pressure-Reducing Systems | 5.2, 5.3, 6.4, 6.6, 7.4, 8.4, 8.5, 8.7 | See Note 6 |
| 62. | Inspection of Air Cleanup Systems | 5.2, 5.3, 10.12 | |
| 63. | Testing of Air Cleanup Systems Components | 5.2, 5.3, 10.12 | |
| 64. | Testing of Air Cleanup Systems | 5.2, 5.3, 10.12 | |
| 65. | Testing of Operational Sequence of Air Cleanup Systems | 5.3, 7.12, 13.4 | |

Notes: 1. The opening phrase in the second sentence of the criterion, "as a minimum," represents an unrealistic extension of today's requirements for the design of engineered safety features. The current design basis accident, which assumes an instantaneous, circumferential rupture of up to the largest pipe in the primary system, represents an extremely conservative design basis. To imply that a break larger than this should be considered is unduly conservative and has very serious implications on plant design. Furthermore, the inclusion of this phrase makes the criterion much less specific and leaves the design basis completely open to interpretation in the future. Because this defeats the whole purpose of the criterion, this phrase is deleted in our interpretation.

2. The criterion requires that two independent failures be considered which may be unduly conservative for some plants. The design of the onsite and offsite power system should be based on an overall availability and reliability analysis of the entire complex of power systems.

   The last sentence of the criterion is therefore interpreted as the following: "The complex of electrical power systems which provide power for engineered safety features shall be designed to meet stated reliability and availability goals. A justification of the bases for selection of these stated goals is required".

3. The design of the engineered safety features should be based on an overall reliability and availability analysis for these features; such an approach will lead to improved safety as it gives proper weighting not only to single failures, but also to combined failures with a high probability of occurrence. The last sentence of the criterion is therefore interpreted as the following: "Engineered safety features shall be designed to meet stated reliability and availability goals. A justification of the bases for selection of these stated goals is required".

Table A.0-7

(Sheet 3)

AEC GENERAL DESIGN CRITERIA - GROUP VII (ENGINEERED SAFETY FEATURES)

(Cont.)
Notes:

4. This criterion is interpreted to reflect the fact that it is a practical impossibility to design each component and each system so that its effectiveness is not impaired by the loss-of-coolant accident. Loss-of-coolant accidents can be postulated in which the component of an emergency core cooling system would be directly involved. The objective for engineered safety features is more properly expressed as follows: "Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss of coolant accident."

5. Item (c) of the criterion, relating to sharing, is confusing and difficult to interpret. Inasmuch as sharing is covered by Criterion 4, and the impairment of these features due to effects of the loss-of-coolant accident is covered under Criterion 42, this entire phrase is unnecessary. The criterion should include a reference to location of pipe breaks as well as to size of break.

6. There is concern about the unavailability of these systems during such a test, particularly since it may be extremely difficult to perform such a test during reactor operation. There is also concern about the design complications which will be required in order to permit such a test. This means adding extra equipment and further complicates an already complex system, which may be detrimental to safety. This criterion is interpreted to mean testing of such systems in subsystems with suitable subsystem analysis, rather than testing of the entire operational sequence.

7. Since the effective functioning of emergency core cooling systems is required to maintain containment integrity, it is not logically consistent to base the design of the containment system on the presumed failure of these same emergency core cooling systems. It is more appropriate to relate the containment design basis directly to the performance capability of the emergency core cooling systems. It appears that the intent of this criterion is to require that the containment systems be designed to handle pressures and temperatures substantially in excess of those which would occur with functioning emergency core cooling systems, as is the case in current licensing practice. This does not appear to bring about any real improvement in overall plant safety. A more modest allowance for any impairment of cooling system performance is, therefore, allowable. The criterion is interpreted as follows:
"The containment structure, including access openings and penetrations and any necessary containment heat removal systems, shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including some allowance for effects from metal-water or other chemical reactions beyond those that would occur with normal operation of emergency core cooling systems at design objection conditions."

8. This broadly applicable and yet very specific requirement is not in keeping with the intent of the general criteria. These requirements should be spelled out in the supplemental criteria is interpreted as the simple statement, "Design of containments shall be in accordance with applicable engineering codes."

9. Leakage rate testing of the containment at design pressure is not now an AEC requirement for plants after they have been placed in service. The severe burden which this would impose has been recognized and modified procedure adopted. This procedure used the relationship between leakage rates measured initially at design pressure, and at some reduced pressure. Such a relationship is then employed to extrapolate subsequent test values for leakage at reduced pressure to the full design pressure of interest.

   This criterion is therefore interpreted as follows: "The containment shall be designed so that integrated leakage rate testing can be done periodically during plant lifetime. Such tests will be made at a pressure which permits extrapolation of results to the design pressure condition, using relationships established initially for comparative leakage at these two conditions."

Table A.0-8

AEC GENERAL DESIGN CRITERIA - GROUP VIII (FUEL AND WASTE STORAGE SYSTEMS)

| Criterion | | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 66. | Prevention of Fuel Storage Criticality | 7.6, 10.2, 10.3 | |
| 67. | Fuel and Waste Storage Decay Heat | 10.5 | |
| 68. | Fuel and Waste Storage Radiation Shielding | 9.1, 9.2, 9.3, 9.4, 10.3, 10.5, 12.2, 12.3 | |
| 69. | Protection Against Radioactivity Release From Spent Fuel and Waste Storage | 5.1-5.3, 9.2, 9.3, 9.4, 10.2, 10.3, 10.5, 12.1, 12.2 | |

Table A.0-9

AEC GENERAL DESIGN CRITERIA - GROUP IX (PLANT EFFLUENTS)

| Criterion | | Conformance (References to Sections of FSAR) | Remarks |
|---|---|---|---|
| 70. | Control of Releases of Radioactivity to the Environment | 1.5, 5.2, 5.3, 7.12, 7.13, 9.2, 9.4, 14.2-14.7 | |