MAY 2 1975

bcc: ACRS (14)
Thomas B. Abernathy, DTIE
J. R. Buchanan, ORNL
A. Rosenthal, ASLAB
N. H. Goodrich, ASLBP

Docket No. 50-324

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Change No. 1
Amendment No. 1
License No. DPR-62

Dear Mr. Jones:

The Nuclear Regulatory Commission has issued Amendment No. 1 to Facility Operating License DPR-62. This amendment consists of the following changes to the Brunswick 2 Technical Specifications: (a) the changes requested in your March 11, 1975 letter; (b) addition of the Decontamination Room to Table 3.5-4 of Appendix B as a monitored release location; and (c) clarification type changes to Sections 3.3 and 4.3 regarding the Rod Sequence Control System.

The Staff Evaluation supporting the changes to the Technical Specifications is enclosed for your information and use. Also enclosed is a copy of a Federal Register Notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by
Walter Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Enclosures:
1. Amendment 1 - DPR-62
2. Staff Evaluation
3. Federal Register Notice

cc: See page 2

Distribution:
Docket File
NRC PDR
Local PDR
LWR 1-2 File
I. Stridiron, ELD
IE (3)
N. Dube(w/o tech specs)
R. C. DeYoung
M. Jinks(2/2 encl)
R. Vollmer
A. Braitman, OAI
R. Powell
M. Maigret
S. Kari(w/o tech specs)
W. Miller:DR:oa(w/o tech specs
M. Grotenhuis, EP-2
P. Kreutzer, EP-2
LWR 1 & 2 BG's
B. Schraf (15 cys)

CP1

OFFICE	L:LWR 1-2	L:LWR 1-2	OELD	L:LWR 1-2	
SURNAME	MMaigret/red	RPowell		WButler	
DATE	5/ /75	5/ /75	5/ /75	5/ /75	

Form AEC-318 (Rev. 9-53) AECM 0240

cc: Richard E. Jones, Esq.
Carolina Power & Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

George F. Trowbridge, Esq.
Shaw, Pittman, Potts & Trowbridge
910 17th Street, N. W.
Washington, D. C. 20006

John J. Burney, Jr., Esq.
Burney, Burney, Sperry & Barefoot
110 North Fifth Avenue
Wilmington, North Carolina 28401

Ms. Janet Godwin, President
Project Environment
202 Bedford Road East
Wilmington, North Carolina 28401

Mr. Dave Hopkins
Environmental Protection Agency
1421 Peachtree Street, N. E.
Atlanta, Georgia 30309

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Mr. W. A. Kopp, Jr.
Chairman, Board of County
Commissioners of Brunswick County
Bolivia, North Carolina 28422

ATTACHMENT TO LICENSE AMENDMENT NO. 1
CHANGE NO. 1 TO THE TECHNICAL SPECIFICATIONS
OPERATING LICENSE NO. DPR-62
Docket No. 50-324

Replace the existing pages of the Technical Specifications listed below with the attached revised pages bearing the same numbers. All changes on these pages are denoted by a marginal line and Amendment No. 1.

Appendix A

(1) Page 1.1-17
(2) Page 3.1-3
(3) Page 3.1-6
(4) Page 3.2-5
(5) Page 3.2-6
(6) Page 3.2-8
(7) Page 3.2-9
(8) Page 3.2-10
(9) Page 3.2-13
(10) Page 3.2-14
(11) Page 3.2-17
(12) Page 3.2-19
(13) Page 3.2-30
(14) Page 3.2-37
(15) Page 3.2-39
(16) Page 3.2-40
(17) Page 3.2-41
(18) Page 3.2-42
(19) Page 3.2-45
(20) Page 3.2-46
(21) Page 3.2-49
(22) Page 3.2-51
(23) Page 3.2-60
(24) Page 3.3-1
(25) Page 3.3-2
(26) Page 3.3-3
(27) Page 3.3-4
(28) Page 3.3-5
(29) Page 3.3-6
(30) Page 3.5-1
(31) Page 3.5-3
(32) Page 3.5-11
(33) Page 3.5-11a
(34) Page 3.6-1
(35) Page 3.6-2
(36) Page 3.6-3
(37) Page 3.6-8
(38) Page 3.7-18
(39) Page 3.7-29
(40) Page 6.5-5
(41) Page 6.5-6
(42) Page 6.5-7

Appendix B

(1) Page 2-29

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CAROLINA POWER AND LIGHT COMPANY
DOCKET NO. 50-324
BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 1
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated March 11, 1975 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by the changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2C(2) of Facility License No. DPR-62 is hereby amended to read as follows:

   "2C(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 1."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Attachment:
Change No. 1
Technical Specifications

Date of Issuance: MAY 2 1975

M 2 1975

SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO DPR-62

CHANGE NO. 1 TO TECHNICAL SPECIFICATIONS

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

Introduction

By letter dated March 11, 1975, Carolina Power & Light Company requested certain changes to the Technical Specifications appended to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2. In addition to the above, certain additional changes have been made by the NRC's Division of Reactor Licensing and agreed to by the licensee to clarify the conditions for operation of Unit 2.

Evaluation

We have reviewed the proposed changes to the Technical Specifications, Appendix A for the Facility Operating License No. DPR-62 as submitted by Carolina Power and Light Company on March 11, 1975 for the Brunswick Steam Electric Plant, Unit 2. In addition, we have included certain changes requested by telephone under our emergency procedures on March 17, 1975, and certain other changes resulting from our review of those matters discussed during our telephone conversation with CP&L officials on April 8, 1975.

The following is an itemized evaluation of each change made in Amendment No. 1.

1. page 1.1-17 - Change 30% to 20% in next to last sentence on page to be consistent with Rod Sequence Control System Section 3.3 of Technical Specifications.
2. page 3.1-3 - Note 14 has been added to APRM High Flux trip function for clarification. Correction has been made to instrument number for High reactor pressure trip instrument numbers.
3. page 3.1-6 - Note 14 was added to indicate that the flow bias scram has a 6 second delay while the 120% does not. This delay of 6 seconds is approximately the time constant of the fuel pins so that the flow bias scram simulates the thermal flux and not the neutron flux.
4. page 3.2-5 - The two changes on this page were to correct instrument numbers.
5. page 3.2-6 - Note 1d added to clarify that mechanical vacuum pump is tripped only on high main steam line radiation signal.
6. page 3.2-8 - Same as 5 above.
7. page 3.2-9 - Trip setting for Steam line high d/p changed to $\leq$ 300% rated flow to assure it is less than the limit used in analysis of the HPCI steam supply line break.
8. page 3.2-10 - Clarification is provided as to action required when minimum conditions are not met. Note (3) added.
9. page 3.2-13 - Same as item 7 above except for RCIC.
10. page 3.2-14 - Same as item 8 above except for Group V Isolation.
11. page 3.2-17 - Trip setting change to units of gallons per minute for clarification ($\leq$53 gpm).

12. page 3.2-19 - Reactor building ventilation is removed under remarks since isolation does not directly occur from low water level signal. Isolation of the reactor building ventilation is initiated from standby gas treatment system contacts.

13. page 3.2-30 - Title of Item 9 of the Table 3.2-8 changed to Containment Pressure Permissive. Delete (-203.5 indicated) which cannot be directly read on scale. Correct typographical error 4-KV under remarks.

14. page 3.2-37 - Item 5 of table 3.2-10 is deleted as it is covered in Specification 3.6.D and does not apply to ADS.

15. page 3.2-39 - Same as 14 above.

16. page 3.2-40 - Correction of typographical error ($\geq$ 3 cps and $\leq$108/125).

17. page 3.2-41 - Correction of typographical errors and clarification by adding note 2 reference.

18. page 3.2-42 - Addition of note 2 to provide clarification regarding minimum number of operable channels requirement.

19. page 3.2-45 - Correction of water level instrument number and group number which provides standby gas treatment initiation and results in reactor building isolation. Correction of minimum number of operable instrument channels for reactor building ventilation item 3 and 4 of table 3.2-12 from 2 to 1. Correction of spelling of annunciate.

20. page 3.2-46 - Correction of reactor low water level instrument number and group number.

21. page 3.2-49 - Correction of instrument number for reactor water level and range of containment radiation.

22. page 3.2-51 - Correction of instrument number for reactor level and containment pressure.

23. page 3.2-60 - Correction of number of radiation monitors provided to initiate isolation of reactor building ventilation.

24. page 3.3-1 - Clarification of when exercising of control drives must be done for surveillance.

25. page 3.3-2 - Section 4.3.A.b was added to provide clarification and requirements for a control rod stuck in the reactor. This section was inadvertently omitted in the initial issued Appendix A.

26. page 3.3-3 - The note at the end of 3.3B was added to indicate that a re-evaluation of the RSCS would be required prior to operation following the first refueling outage.

27. page 3.3-4 - The words and bypassed were deleted at end of first sentence of 3.3.B.3d since 3.3.B.6 was added for clarification. Section 4.3.B.3f was revised to clarify that RSCS consists of two parts the sequence portion and the group notch portion.

28. page 3.3-5 - Section 3.3.B.6 and 4.3.B.6 were added to clarify the relaxation of the RSCS restraints and surveillance requirements. These sections were inadvertently omitted in the initial issuance of the Appendix A.

29. page 3.3-6 - Section 4.3.C.2 was revised to clarify the surveillance requirements for measurement of control rod scram times.

30. page 3.5-1 - Item (a) and (d) under 4.5A.1 frequency were inadvertently reversed in the initial issuance of Appendix A.

31. page 3.5-3 - Typographical error correction of system pressure against which system pumps must deliver 17,000 gpm.

32. page 3.5-11 - Under 3.5.H and 4.5.H clarification has been added to consider condition that the automatic fill station is inoperable and states the necessary surveillance to assure piping is water filled to prevent water hammer. Typographical error correction 3.5.G under Surveillance is corrected to 4.5.G.

33. page 3.6-1 - Clarification of Section 3.6A.2 is provided to permit relaxation of reactor vessel shell and fluid temperatures when the reactor vessel is vented.

34. page 3.6-2 - Section 3.6.A2 correction same as item 33 above. Footnote at bottom of page correction of I-132 and I-133 typographical error.

35. page 3.6-3 - Section 3.6B.2. Typographical error per hour not per hours.

36. page 3.6-8 - Correction of Section number 4.6.G under Surveillance.

37. page 3.7-18 - Typographical error of maximum operating time for RHR discharge isolation valves to radwastes. Closing time should have been 30 seconds, the same as other RHR isolation valves.

38. page 3.7-29 - Typing of 10 CFR 100 corrected.

39. page 6.5.-5 - Correction of spelling (nuclear).

40. page 6.5-6 - Typographical error Section 50-59. A change from plant managers to supervisory staff under 6.5.2.8 Quorum was made to provide additional flexibility since it was not intended to be only plant managers.

41. page 6.5-7 - Typographical error: The word "with" changed to "within".

42. page 2-29 of Appendix B. This change was inadvertently omitted in the initial issued Appendix B. It requires monitoring of the effluent from the Decontamination room.

These changes have been reviewed, discussed and found acceptable to both the NRC staff and the licensee. All the changes proposed for Amendment No. 1 to DPR-62 can be classified in one of the following categories:

1. Typographical errors;
2. Changes to provide for consistency throughout the Technical Specifications;
3. Clarification of the intent of the Technical Specifications; and
4. Additional information for completeness of understanding and for agreement with as built conditions, which we find acceptable.

Our review and evaluation of all the changes included in this Amendment No. 1, Change No. 1 indicates that the changes do not involve significant hazards considerations or an unreviewed safety matter.

Conclusion

Based upon our review of the changes to the Technical Specifications: (a) proposed by Carolina Power and Light Company (CP&L) in its letter dated March 11, 1975; and (b) initiated by the NRC staff and agreed to by the licensee, we find that none of these changes involves a significant hazards consideration, or involves an unreviewed safety matter. We have concluded for each of the above identified changes, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequence of accidents previously considered, it does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation with the identified changes; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this Amendment No. 1 will not be inimical to the common defense and security or to the health and safety of the public.

UNI[illegible] STATES NUCLEAR REGULATORY COM[illegible]SION

DOCKET NO. 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 1 to Facility Operating License No. DPR-62, which was issued to Carolina Power & Light Company on December 27, 1974. Amendment No. 1 to DPR-62 revises the Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit 2, located on the Cape Fear River, near Southport in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

The purpose of this amendment to the Technical Specifications is to clarify certain provisions and to correct certain typographical errors.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (The Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see: (1) the application for amendment, dated March 11, 1975; (2) Amendment No. 1 to License No. DPR-62, with Change No. 1; and (3) the Commission's related Safety Evaluation.

All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Southport - Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

A copy of Items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 2nd day of May, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

BASES:

2.1 LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

by manual operator action using a switch on the R-T-G board. The assignment of control rods to the select rod insert function is based on the startup and fuel warranty service associated with each control rod pattern, on RSCS considerations, and a dynamic function of both time and core patterns.

Approximately ten percent of the control rods in the reactor will be assigned to the select rod insert function by the operator. This selection will be accomplished by moving the rod scram test switch for those rods from the "NORMAL" position to the "SELECT ROD INSERT" position.

For Brunswick Unit No. 2, loss of turbine control valve hydraulic pressure will initiate the select rod insert function and the pre-selected group of control rods will be fully inserted. A reactor protection system trip will not be initiated prior to determination of turbine bypass valve status. Determination of the bypass valve status will be delayed by 200 msec referenced to the start of the low turbine control valve hydraulic pressure signal. If the bypass valves are not open, as determined by limit switches, the reactor protection system will scram the reactor. Any rod selected for Select Rod Insert should also have other rods in its notch group selected to ensure that the RSCS (Rod Sequence Control System) criteria of plus-minus one notch position equality is met when the rod pattern is between 50% rod density and 20% reactor power. It is possible that a rod pattern within these limits may occur after the Select Rod Insert Function operates.

Amendment 1

BASES:

2.1 LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

F & G. Main Steamline Isolation on Low Pressure and Main Steamline Isolation Scram

The low pressure isolation of the main steamlines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steamline isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steamline low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure.

H & I. Reactor Low Water Level Setpoints for Initiation of HPCI and RCIC, Automatic Depressurization, and Starting LPCI and Core Spray Pumps

These systems maintain adequate cooling inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses reported in FSAR Section 14 demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Trip Settings	Modes in Which Functions Must be Operable: Refuel (1)	Startup	Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3)
1. Mode switch C72A-S1		X	X	X	1	A
2. Manual trip C72A-S3A, B		X	X	X	1	A
3. IRM						
High flux	≤120/125 of scale	X	X	(12)	3	A
Inoperative		X	X	(12)	3	A
4. APRM						
High Flux (4,14) (flow bias)	$\leq(0.66W+54)\frac{(2.60)}{T.P.F.}$			X	2	B
(fixed)(14)	≤120% of rated power			X	2	B
Inoperative		X		X	2(5)	B
Downscale	≥3/125 of scale			X(13)	2	B
Startup	≤15% of rated power	X	X		2	A
5. High reactor pressure B21-PS-N023A, B,C,D	≤1045 psig	X(10)	X	X	2	A
6. High drywell pressure C72-PS-N002A, B,C,D	≤2 psig	X(11)	X(11)	X	2	A
7. Reactor low water level #1 B21-LIS-N017A, B,C,D	≥12.5 inch (6)	X	X	X	2	A

Amendment #1

Amendment #1

BSEP 1 & 2

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must be Operable: Refuel (1)	Startup	Run	Min. No. Operable Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions for Operation Are Not Satisfied (3)
8. Scram discharge volume high level C11/C12-LSH-N013A,B,C,D	≤ 109 Gallons	X	X	X	2	A
9. Main steamline high radiation D12-RM-K603A,B,C,D	≤ 3x normal background at rated power	X	X	X	2	C
10. Main steamline isolation valve closure B21-ZS-F022A,B,C,D B21-ZS-F028A,B,C,D	≤ 10% valve closure (7)	X	X	X	4	C
11. Turbine stop valve closure EHC-SVOS-1,2,3,4	≤ 10% valve (9) closure			X	4	D

BSEP-1 & 2

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must be Operable: Refuel (1)	Startup	Run	Min No. Operable Instrument Channels Per Trip System (2)	Required Conditions When When Minimum Conditions for Operations are Not Satisfied (3)
First stage turbine pressure permissive C71-PS-N003A,B,C,D	(9)			X	2	D
12. Turbine control valve fast closure EHC-PSL-1756 EHC-PSL-1757 EHC-PSL-1758 EHC-PSL-1759	≥ 850 psig control oil pressure (8)	X	X	X	2	D

NOTES:

(1) When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

A. Mode switch in SHUTDOWN

B. Manual scram

C. High flux IRM

D. Scram discharge volume high water level

It is possible during reactor operation to switch to the refuel mode and remain critical. The requirement to have all other scram functions operable in the refuel mode is therefore to assure that shifting to this mode during reactor operation does not diminish the protection afforded by the RPS.

(2) There shall be two operable, one operable and one tripped, or two tripped trip systems for each function.

BSEP-1 & 2

NOTES (Cont'd)

(3) When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met, the appropriate actions listed below shall be taken:

A. Initiate insertion of operable rods and complete insertion of all operable rods within eight hours.

B. Reduce power level to IRM range and place mode switch in the STARTUP position within eight hours.

C. Reduce turbine load and close main steam line isolation valves within eight hours.

D. Reduce reactor power to less than 30% of rated within eight hours.

(4) "W" is the reactor driving loop flow in percent of rated (see Specification 2.1.A.1).

(5) To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 11 LPRM inputs.

(6) Twelve and one half inches on the water level instrumentation is 177 inches above the top of the active fuel.

(7) A main steam isolation valve closure bypass is permitted when the reactor mode switch is in either the SHUTDOWN, REFUEL, or STARTUP position.

(8) For Unit 2, low control oil pressure initiates select rod insert but shall not initiate reactor protection system trip before determination of turbine bypass valve status. In both Units, this scram is bypassed if the first stage turbine pressure is less than 30 percent of normal rated power.

(9) A turbine stop valve closure bypass is permitted when the first stage turbine pressure is less than 30 percent of normal rated power.

(10) Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

(11) Not required to be operable when the primary containment integrity is not required.

(12) IRM's are bypassed when APRM's are on scale and the reactor mode switch is in the RUN position.

(13) The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and $\leq$ 120/125 of full scale. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

(14) The APRM high flux signal is fed through a time constant curcuit of approximately 6 seconds. The APRM fixed scram does not incorporate the time constant, but responds directly to instantaneous neutron flux.

BSEP-1 & 2

TABLE 3.2-1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

GROUP I ISOLATION (1)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Condition for Operation are not Satisfied	Remarks
1.	Reactor low water level #2 B21-LIS-N024A,B B21-LIS-N025A,B	≥ - 38" indicated level	Two	(2.a.)	
2.	Steamline area high temperature B21-TS-N010A,B,C,D B21-TS-N011A,B,C,D B21-TS-N012A,B,C,D B21-TS-N013A,B,C,D	≤ 200 F	Two of four in each of two channels	(2.b.)	
3.	Steamline high flow B21-dPIS-N006A,B,C,D B21-dPIS-N007A,B,C,D B21-dPIS-N008A,B,C,D B21-dPIS-N009A,B,C,D	≤ 140% of rated flow	Two per steamline	(2.b.)	
4.	Main steamline low pressure B21-PS-N015A,B,C,D	≥ 850 psig	Two (3)	(2.b.)	
5.	Main steamline high radiation D12-RM-K603A,B,C,D	≤ 3 x background at rated power	Two (2)	(2.b.)	Has contacts in reactor protection system
6.	Steamline high flow while in STARTUP B21-dPIS-N006A B21-dPIS-N007B B21-dPIS-N008C B21-dPIS-N009D	≤ 40% of rated flow	Two (4)	(2.b.)	

Amendment #1

Amendment #1

BSEP-1 & 2

TABLE 3.2-1

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

GROUP I ISOLATION (1) (Cont'd)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Condition for Operation are not Satisfied	Remarks
7.	Low condenser vacuum B21-PS-N056A,B,C,D	>18" Hg.	Two	(2.b.)	
8.	Turbine building area high temperature B21-TS-3225A,B,C,D B21-TS-3226A,B,C,D B21-TS-3227A,B,C,D B21-TS-3228A,B,C,D B21-TS-3229A,B,C,D B21-TS-3230A,B,C,D B21-TS-3231A,B,C,D B21-TS-3232A,B,C,D	$\leq 200^{o}F$	Two of four in each of four channels	(2.b.)	

BSEP-1 & 2

NOTES:

(1) Group I isolation includes:

a. Main steamline isolation valves
b. Main steamline drain valves
c. Reactor water sample valves (only on low water level #2 or high main steam line radiation signal)
d. Mechanical vacuum pump trip (only on high main steamline radiation signal)

(2) If the minimum number of operable instrument channels is not available for one trip system, that trip system shall be tripped. If the minimum number of operable of tripped instrument channels is not available for both trip systems, the appropriate actions listed below shall be taken:

a. Initiate an orderly shutdown and have reactor in the cold shutdown condition in 24 hours.
b. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

(3) The main steamline low pressure need be available only in the RUN mode.

(4) Not required in RUN mode. Applies to Unit 2 only.

Amendment #1

TABLE 4.2-1

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP 1 ISOLATION

	Trip Function	Functional Test	Calibration	Instrument Check
1.	Reactor low water level #2 B21-LIS-N024A,B B21-LIS-N025A,B	once/month	(1)	once/day
2.	Steamline area high temperature B21-TS-N010A,B,C,D B21-TS-N011A,B,C,D B21-TS-N012A,B,C,D B21-TS-N013A,B,C,D	once/month without setpoint vertification	once/operating cycle	once/day
3.	Steamline high flow B21-dPIS-N006A,B,C,D B21-dPIS-N007A,B,C,D B21-dPIS-N008A,B,C,D B21-dPIS-N009A,B,C,D	once/month	(1)	N/A
4.	Main steamline low pressure B21-PS-N015A,B,C,D	once/month	(1)	N/A
5.	Main steamline high radiation D12-RM-K603A,B,C,D	check with reactor protection system	once/operating cycle	once/shift
6.	Steamline high flow while in STARTUP (2) MODE B21-dPIS-N006A B21-dPIS-N007B B21-dPIS-N008C B21-dPIS-N009D	once/month	(1)	N/A

BSEP-1 & 2

TABLE 4.2-1

MINIMUM TEST & CALIBRATION FREQUENCIES
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP 1 ISOLATION (Cont'd)

	Trip Function	Functional Test	Calibration	Instrument Check
7.	Low condenser vacuum B21-PS-N056A,B,C,D	once/month	(1)	N/A
8.	Turbine building area high temperature B21-TS-3225A,B,C,D B21-TS-3226A,B,C,D B21-TS-3227A,B,C,D B21-TS-3228A,B,C,D B21-TS-3229A,B,C,D B21-TS-3230A,B,C,D B21-TS-3231A,B,C,D B21-TS-3232A,B,C,D	once/month without setpoint verification	once/operating cycle	once/day

Group I - Isolation logic system functional test to be performed once/operating cycle.

Group I - Isolation included:

a. Main steamline isolation valves.
b. Main steamline drain valves.
c. Reactor water sample valves (low water level #2 or high main steam line radiation).
d. Mechanical vacuum pump trip (only on high main steamline radiation signal).

NOTES:

(1) When a functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

(2) Applies to Unit 2 only.

BSEP-1 & 2

Amendment #1

TABLE 3.2-2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
HIGH PRESSURE COOLANT INJECTION SYSTEM
GROUP IV ISOLATION (2)

	Trip Function	Trip Setting	Minimum number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Met	Remarks
1.	Steamline high d/p (steamline break) E41-dPIS-N004 E41-dPIS-N005	≤300% rated flow	2	(1)	
2.	HPCI turbine steam-line low pressure E41-PSL-N001A,B,C,D	100 (± 3) psig	2	(1)	
3.	HPCI turbine exhaust diaphragm high pressure E41-PSH-N012A,B,C,D	10 (± 0.7) psig	2	(1)	
4.	Suppression pool high ambient temperature E51-TS-N603C,D	≤200F	2	(1)	
5.	Suppression pool area vent inlet/outlet high differential temperature E51-dTS-N604C,D	≤50F	2	(1)	
6.	Emergency area cooler high temperature E41-TS-N602A,B	≤175F	2	(1)	

Amendment #1

BSEP-1 & 2

TABLE 3.2-2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
HIGH PRESSURE COOLANT INJECTION SYSTEM
GROUP IV ISOLATION (2) (Cont'd)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are Not Met	Remarks
7.	HPCI equipment room vent inlet/outlet high differential temperature E41-dTS-N601A,B	≤ 50F	2	(1)	
8.	HPCI equipment room main steamline area high temperature E41-TS-3314 E41-TS-3315 E41-TS-3316 E41-TS-3317 E41-TS-3318 E41-TS-3354	≤ 175F	2	(1)	
9.	Bus power monitor E41-K55 E41-K56	NA	1	(3)	Annunciate only

NOTES:

(1) Close isolation valves in system and comply with Specification 3.5

(2) Group IV isolation includes:

a. HPCI inboard steam isolation valve
b. HPCI outboard steam isolation valve
c. HPCI torus suction valve (does not close on HPCI equipment room main steam line area high temperature)

(3) Monitor bus power daily and comply with Specification 3.5 if power is lost.

MARCH 1975

BSEP-1 & 2 Amendment #1

Amendment #1

TABLE 3.2-3

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

REACTOR CORE ISOLATION COOLING SYSTEM

GROUP V ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditons for Operation are not Met	Remarks
1.	RCIC steamline high d/p (steamline break) E51-dPIS-N017 E51-dPIS-N018	≤300% rated flow	2	(1)	
2.	RCIC turbine steamline low pressure E51-PS-N019A,B,C,D	50 (± 2) psig	2	(1)	
3.	RCIC turbine high exhaust diaphragm pressure, E51-PS-N012A,B,C,D	10 (± 4) psig	2	(1)	
4.	Suppression pool high ambient temperature E51-TS-N603A,B	≤ 200F	2	(1)	
5.	Suppression pool area vent inlet/outlet high differential temperature. E51-dTS-N604A,B	≤ 50F	2	(1)	
6.	RCIC equipment room high ambient temperature, E51-TS-N602A,B	≤ 175F	2	(1)	
7.	RCIC equipment room vent inlet/outlet high differential temperature E51-dTS-N601A,B	≤ 50F	2	(1)	

BSF 1 & 2
Amendment #1

TABLE 3.2-3 (Cont'd)

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
REACTOR CORE ISOLATION COOLING SYSTEM
GROUP V ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Met	Remarks
8.	RCIC equipment room main steamline area high temperature E51-TS-3319 E51-TS-3320 E51-TS-3321 E51-TS-3322 E51-TS-3323 E51-TS-3355	≤ 175F	2	(1)	
9.	Bus power monitor E51-K42 E51-K43	N/A	1	(3)	Annunciate only

NOTES: (1) Close isolation valve in system and comply with Specification 3.5.

(2) Group V includes

a. RCIC inboard steam isolation valve.
b. RCIC outboard steam isolation valve.

(3) Monitor bus power daily and comply with Specification 3.5 if power is lost.

BSEP-1 & 2
Amendment #1

Amendment #1

TABLE 3.2-4

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

REACTOR WATER CLEANUP SYSTEM

GROUP III ISOLATION (2)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are Not Met	Remarks
1.	Reactor low level #1 B21-LIS-N017A,B,C,D	≥12.5" indicated level	2	(1)	Has contacts in Group II, Reactor Building ventilation isolation and SGTS initiation systems.
2.	Reactor water cleanup high temperature G31-TIS-N008	≤140°F	1	(1)	
3.	Reactor water cleanup high differential flow G31-dFS-N603	≤ 53 gpm	1	(1)	
4.	Standby liquid control system initiated C41-RMS-S1A,B	N/A	N/A	(1)	
5.	Reactor water cleanup space high temperature G31-TS-N600A,B,C,D,E,F	100 - 150°F	2	(1)	
6.	Vent air inlet/outlet high differential temperature G31-dTS-602A,B,C,D,E,F	≤ 50°F	2	(1)	

NOTES:

(1) Close isolation valves in cleanup system and comply with Specification 3.6.B.

(2) Group III isolation includes:
   a. RWCU outboard isolation valve
   b. RWCU inboard isolation valve (does not close on SLC initiation or RWCU high temperature).

BSEP-1 & 2
Amendment #1

TABLE 4.2-4

MINIMUM TEST & CALIBRATION FREQUENCIES

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

REACTOR WATER CLEANUP SYSTEM

GROUP III ISOLATION (2)

	Trip Function	Functional Test	Calibration	Instrument Check
1.	Reactor low water level #1 B21-LIS-N017A,B,C,D	This reactor low water level #1 switch is on the same instrument as the PCIS low water level #1 switch and will be functionally tested and calibrated at the same time.		
2.	Reactor water cleanup high temperature G31-TIS-N008	once/month	(1)	NA
3.	RWCU high differential flow G31-dFS-N603	once/month	(1)	once/day
4.	Standby liquid control system initiated G41-RMS-S1A,B	once/operating cycle	N/A	N/A
5.	RWCU space high temperature G31-TS-N600A,B,C,D,E,F	once/month	(1)	N/A
6.	RWCU vent air inlet/ outlet high differential temperature G31-dTS-602A,B,C,D,E,F	once/month	(1)	N/A.

RWCU isolation logic system functional test will be performed once/6 months

NOTES:

(1) When a functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

(2) Group III isolation includes:

a. RWCU outboard isolation valve
b. RWCU inboard isolation valve (does not close on SLC initiation or RWCU high temperature).

BSEP-1 & 2

TABLE 3.2-5
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP II ISOLATION (2)

Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels Per Trip System	Required Action When Minimum Condition for Operation Is Not Satisfied	Remarks
1. High drywell pressure C72-PS-N002A,B,C,D	≤2 psig	2	(1)	
2. Reactor low water level #1 B21-LIS-N017A,B,C,D	≥+ 12.5" indicated	2	(1)	Initiates Group II isolation and has contacts in cleanup isolation, and SGTS initiation.

Amendment #1

BSEP-1 & 2

NOTES:

(1) Close isolation valves in system and comply with Specification K3.5.

(2) Group II Isolation includes:
- a. Drywell equipment drain discharge valves
- b. Drywell floor drain discharge valves
- c. Reactor head spray isolation valves (4)
- d. Shutdown cooling suction valves (4)
- e. RHR system radwaste discharge isolation valves
- f. RHR process sampling valves
- g. Nitrogen makeup and inerting inlet valve (3)
- h. Suppression chamber inerting inlet valve (3)
- i. Drywell inerting inlet valve (3)
- j. Suppression chamber purge exhaust isolation valve (3)
- k. Suppression chamber vent valve bypass valve (3)
- l. Drywell purge exhaust isolation valve (3)
- m. Drywell purge exhaust backup valve (3)
- n. Containment air purge isolation valve (3)
- o. Suppression chamber vent valve (3)
- p. Drywell purge exhaust backup valve bypass valve (3)
- q. Suppression chamber makeup and containment atmosphere dilution inlet valve (3)

TABLE 3.2-5
PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION
GROUP II ISOLATION (2) (Cont'd)

NOTES:

(2) Group II isolation (Cont'd)

r. Drywell makeup and containment atmosphere dilution inlet valve (3)
s. Drywell vent isolation valve (3)
t. Drywell vent backup valve (3)
u. Containment atmosphere dilution inlet valve (3)
v. Containment atmosphere dilution inlet bypass valve (3)
w. Traveling in-core probe isolation valves
x. RHR inboard injection valves (The signal only applies if the RHR system is in the shutdown cooling mode)

(3) Also isolates on high radiation signal.

(4) Also isolates on high reactor pressure signal.

BSEP-1 & 2

TABLE 3.2-8

INSTRUMENTATION THAT INITIATES OR CONTROLS
THE CORE AND CONTAINMENT COOLING SYSTEMS

LOW PRESSURE COOLANT INJECTION SYSTEM A & B (1)

	Trip Function	Trip Level Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Satisfied	Remarks
1.	High Drywell pressure E11-PS-N011A,B,C,D	≤ 2 psig	2	(2)	Initiates LPCI and has contacts in core spray, HPCI, ADS, and diesel start
2.	Reactor low water level #3 B21-LIS-N031A,B,C,D	≥ 17" above TAF (-147.5" instrument)	2	(2)	Initiates LPCI and has contacts in core spray, ADS, and diesel start
3.	Low reactor pressure B21-PS-N021A,B,C,D	300-350 psig	2	(2)	Permissive for opening LPCI injection valve, closing permissive for recirculation discharge valves
4.	LPCI pump start time E21-K2A,B	9 ≤ t ≤ 11 sec	1	(3)	In conjunction with loss of power initiate sequential starting of CSCS pumps
5.	LPCI pump discharge pressure interlock E11-PS-N020A,B,C,D	100 (± 9) psig	2	(3)	Prevents ADS actuation pending confirmation of LPCI or CS pump running interlock

BSEP-1 & 2

TABLE 3.2-8 (Cont'd)

	Trip Function	Trip Level Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Satisfied	Remarks
6.	RHR (LPCI) trip system bus power monitor E11-K106A,B	NA	1	(2)	Monitors availability of power to logic system
7.	Startup transformer under voltage 27/59 S	≥ 75% of tap setting nominal voltage	1		Energizes offsite power monitor relay which initiates diesel loading sequence for onsite emergency power.
8.	Emergency bus power monitor relay E11-K2A,B E11-K3A,B	NA	1		Permit starting of RHR 4-kv motor
9.	Containment Pressure Permissive E11-PS-N019A,B,C,D	2.5(±0.5) psig	2	(2)	Prevents inadvertent operation of containment spray during accident condition
10.	Reactor low level inside shroud B21-LITS-N036 B21-LITS-N037	≥39" below TAF (2/3 core height)	1		Prevents inadvertent operation of containment spray during accident condition

TABLE 3.2-10

INSTRUMENTATION THAT INITIATES OR CONTROLS
THE CORE AND CONTAINMENT COOLING SYSTEMS
AUTOMATIC DEPRESSURIZATION SYSTEM

	Trip Function	Trip Level Setting	Minimum Number of Operable Instrument Channels per Trip System (3)	Required Action When Minimum Conditions for Operation are Not Satisfied	Remarks
1.	High drywell pressure E11-PS-N010A,B,C,D	≤ 2 psig	2	(1)	Initiates ADS (2) and has contacts in LPCI, core spray, HPCI and diesel start
2.	Reactor low water level #3 B21-LIS-N031A,B,C,D	≥ 17" above TAF (-147.5 instrument)	2	(1)	Initiates ADS (2) and has contacts in LPCI, core spray, HPCI and diesel start
3.	Time delay timer B21-TDPU-K5A,B	≤ 120 sec	1	(1)	In conjunction with reactor low water level #3, high drywell pressure, and LPCI or core spray pump running interlock, initiates auto blowdown
4.	ADS trip system bus power monitor B21-K1A,B	N/A	1	(1)	Monitors availability of power to logic and valves

BSEP-1 & 2

Amendment #1

TABLE 3.2-10 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS
THE CORE AND CONTAINMENT COOLING SYSTEMS
AUTOMATIC DEPRESSURIZATION SYSTEM

NOTES:

(1) Any one of the two trip systems will initiate ADS. If the minimum number of operable channels in any one trip system is not available, the requirements of Specification 3.5 must be met.

(2) ADS is initiated on the following conditions existing simultaneously: high drywell pressure, reactor low water level #3, time delay of 120 sec. and, one LPCI or core spray pump running.

(3) There shall be two operable, one operable and one tripped, or two tripped trip systems for each trip function. If the minimum number of operable instrument channels is not available for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately. If the condition lasts longer than seven days, both trip systems shall be tripped. If the minimum number of instrument channels is not available for both trip systems, both trip systems shall be tripped.

BSEP-1 & 2

TABLE 4.2-10

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS
AUTOMATIC DEPRESSURIZATION SYSTEM

	Trip Function	Functional Test	Calibration	Instrument Check
1.	High drywell pressure E11-PS-N010A,B,C,D	once/month	(1)	N/A
2.	Reactor low water level #3 B21-LIS-N031A,B,C,D	Same level instruments that are used for core spray initiation due to reactor low water level #3 and will be functionally tested and calibrated at same time		
3.	Time delay time B21-TDPU-K5A,B	once/operating cycle	(1)	N/A
4.	ADS trip system bus power monitor B21-K1A,B	once/month	N/A	N/A

ADS subsystem logic system functional test will be performed once/6 months cycle.

NOTES:

(1) When functional test shows the setpoints are out of specified limits, a calibration will be performed immediately.

TABLE 3.2-11

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels (2)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
1. Startup range monitor						
a. Upscale SRM channels A,B,C,D Relay C51-K4	3	X	X		$\leq 10^5$ cps	Bypass if mode switch in RUN or when IRM range switch on RANGE 8 or above.
b. Inoperative SRM channels A,B,C,D Relay C51-K1	3	X	X		(1)	Bypass if mode switch in RUN or when IRM range switch on RANGE 8 or above.
c. Downscale SRM channels A,B,C,D Relay C51-K2	3	X	X		$\geq$ 3 cps	Bypass if mode switch in RUN or when IRM range switch on RANGES 3 or above.
d. Detector not in startup position SRM channels A,B,C,D Relay C51-K9A,B,C,D	3	X	X		Detector motor module unit switch LS-4 not closed (detector not full in)	Bypassed when the count rate is $\geq$ 100 cps IRM on RANGES 3 or above.
2. Intermediate range monitor						
a. Upscale IRM channels A through H, Relay C51-K52	6	X	X		$\leq$ 108/125 full scale	Bypassed in run mode.
b. Inoperative IRM channels A through H Relay C51-K54	6	X	X		(1)	Bypassed in run mode.

Amendment #1

BSEP-1 & 2

Amendment #1

Amendment #1

TABLE 3.2-11 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels (2)	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
c. Detector not in "full in" position, channels A through H, Relays C51-K9E through H, & J through M	6	X	X		Detector motor module limit switch LS-4 not closed (detector not full in)	Bypassed in run mode.
d. Downscale IRM channels A through H, Relay C51-K51	6	X	X		$\geq$ 3/125 of Scale	Bypassed in run mode and when IRM is in RANGE 1.
3. Average power range monitor						
a. Upscale APRM channels A through F, Relays K1 & K7	4			X	$\leq (0.66W+42) \frac{2.60}{T.P.F.}$	
b. Inoperative APRM channels A through F Relays K2 & K8	4	X	X	X	(1)	
c. Downscale APRM channels A through F Relays K3 & K9	4			X	$\geq$ 3/125 of Full Scale	Only active when mode switch is in RUN
d. Upscale startup APRM channels A through F Relay K18	4	X	X		$\leq$ 12% power	Bypassed when in run mode.

Amendment #1 Amendment #1 Amendment #1 Amendment #1

SEP-1 & 2

TABLE 3.2-11 (Cont'd)

CONTROL ROD BLOCKS INITIATED FROM NEUTRON MONITORING SYSTEM

Trip Function	Minimum Number of Operable Instrument Channels	Modes in Which Function Must Be Operable: Refuel	Startup	Run	Trip Setting	Remarks
4. Rod block monitor						
a. Upscale RBM channels A,B Relay K1	1			X	$\leq (0.66W+42)\ \frac{2.60}{T.P.F.}$	
b. Downscale RBM channels A,B Relay K2	1			X	$\geq$ 3/125 of full scale	Only active when mode switch is in RUN **and reactor power is $\geq$ 30%**
c. Inoperative RBM channels A,B Relay K3				X	(1)	

NOTES:

(1) The inoperative trips are produced by the following conditions:

(a) SRM and IRM

1) Mode switch not in OPERATE
2) High voltage power supply voltage low
3) Circuit boards not in circuit

(b) APRM

1) Mode switch not in OPERATE
2) Less than 11 LPRM inputs
3) Circuit boards not in circuit

(c) RBM

1) Mode switch not in OPERATE
2) Circuit boards not in circuit
3) RBM fails to null
4) Less than required number of LPRM inputs for rod selected.

(2) If the minimum number of channels cannot be met for one out of two trip systems, seven days is allowed before requiring the affected trip system to be tripped. If both trip systems do not meet the minimum number of operable channels for operation, both trip systems shall be tripped.

Amendment #1

BSEP-1 & 2

TABLE 3.2-12

REACTOR BUILDING VENTILATION SYSTEM ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION (1)

	Trip Function	Trip Setting	Minimum Number of Operable Instrument Channels per Trip System	Required Action When Minimum Conditions for Operation are not Met	Remarks
1.	Reactor Low Water Level #1 B21-LIS-N017A,B,C,D	≥ 12.5" indicated	2	(2)	Has contact in Group II isolation
2.	High Drywell Pressure C72-PS-N002A,B,C,D	≤ 2 psig	2	(2)	Has contact in Group II isolation
3.	Reactor Building Ventilation Monitors D12-RM-N010A,B	Upscale ≤ 11 mr/hr	1		
4.	Reactor Building Ventilation Monitors, D12-RM-N010A,B	Downscale	1		Annunciate instrument failure in control room

NOTES:

(1) a. Start standby gas treatment
b. Close inboard primary containment purge and vent valves
c. Shutdown and isolate Reactor Building vent system

(2) If the minimum number of operable instrument channels is not available in either trip system for more than 24 hours, the Reactor Building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

BSEP-1 & 2 Amendment #1 Amendment #1 Amendment #1

TABLE 4.2-12

MINIMUM TEST & CALIBRATION FREQUENCIES

REACTOR BUILDING VENTILATION SYSTEM ISOLATION & STANDBY GAS TREATMENT SYSTEM INITIATION

	Trip Function	Functional Test	Calibration	Instrument Check
1.	Reactor low water level #1 B21-LIS-N017A,B,C,D	This reactor low water level #1 switch is on the same instrument as the PCIS Group II reactor low water level #1 switch and will be functionally tested and calibrated at the same time.		
2.	High drywell pressure C72-PS-N002A,B,C,D	This high drywell pressure switch is on the same instrument as the reactor protection system high drywell pressure switch and will be functionally tested and calibrated at the same time.		
3.	Reactor Bldg. ventilation monitors (upscale) D12-RM-N010A,B	once/month	once/operating cycle	once/day
4.	Reactor Bldg. ventilation monitors (downscale) D12-RM-N010A,B	once/3 months	once/operating cycle	once/day

Logic System Functional Test	Frequency
a. Reactor Building isolation	once/6 months
b. Standby gas treatment system actuation	once/6 months

c. Primary containment purge and vent isolation logic functional test will be performed as part of the Group II isolation logic functional test.

Amendment #1

BSEP-1 & 2

TABLE 3.2-14

POST ACCIDENT MONITORING INSTRUMENTATION

	Instrument	Minimumu No. of Operable Instrument Channels	Instrument No.	Type and Indication and Range	Notes
1.	Reactor water level	2	B21-LI-R604 A,B B21-LR-R615	Indicator -150/0/+60" water Recorder -100" to +200" water	(1) (2)
2.	Reactor pressure	2	B21-PI-R004 A,B C32-LPR-R608	Indicator 0-1500 psig Recorder 0-1200 psig	(1) (2)
3.	Containment pressure	2	CAC-PI-2599 CAC-PR-1257	Indicator 0-75 psig Recorder 0-75 psig	(1) (2)
4.	Containment temperature	2	CAC-TR-1258 C91-P602	Recorder 0-400 F Computer 0-400 F	(1) (2)
5.	Suppression chamber atmosphere temperature	2	CAC-TR-1258 C91-P602	Recorder 0-400 F Computer 0-400 F	(1) (2)
6.	Suppression chamber water level	2	CAC-LI-2601 CAC-LR-2602	Indicator -6 to 6' Recorder -6 to +6'	(1) (2)
7.	Suppression chamber water temperature	2	CAC-TR-1258 C91-P602	Recorder 0-400 F Computer 0-400 F	(1) (2)
8.	Containment radiation	2	CAC-AR-1260 CAC-AR-1261	Recorder 10^1 to 10^6 cpm Recorder 10^1 to 10^6 cpm	(1) (2)
9	Containment oxygen	2	CAC-AR-1259 CAC-AR-1263	Recorder/Indicator 0-5%, 0-25% Recorder/Indicator 0-5%, 0-25%	(1) (2)
10.	Containment hydrogen	2	CAC-AR-1259 CAC-AR-1263	Recorder/Indicator 0-10%, 0-20% Recorder/Indicator 0-10%, 0-20%	(1) (2)

Amendment #1

Amendment #1

BSEP-1 & 2

TABLE 3.2-14 (Cont'd)

NOTES:

(1) From the date one of these parameters is reduced to one indication, continued operation is permissible during the succeeding 30 days unless such instrumentation is made operable sooner.

(2) If the requirements of note (1) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

TABLE 4.2-14

MINIMUM TEST AND CALIBRATION FREQUENCY FOR POST ACCIDENT MONITORING INSTRUMENTATION

	Instrument Channel	Instrument Numbers	Calibration Frequency	Instrument Check
1.	Reactor level	B21-LI-R604A,B B21-LR-R615	Once/6 months	Each shift
2.	Reactor pressure	B21-PI-R004A,B C32-LPR-R608	Once/6 months	Each shift
3.	Containment pressure	CAC-PI-2599 CAC-PR-1257	Once/6 months	Each shift
4.	Containment temperature	CAC-TR-1258 C91-P602	Once/6 months	Each shift
5.	Suppression chamber atmosphere temperature	CAC-TR-1258 C91-P602	Once/6 months	Each shift
6.	Suppression chamber water level	CAC-LI-2601 CAC-LR-2602	Once/6 months	Each shift
7.	Suppression chamber water temperature	CAC-TR-1258 C91-P602	Once/6 months	Each shift
8.	Containment radiation	CAC-AR-1260 CAC-AR-1261	Once/6 months	Each shift
9.	Containment oxygen	CAC-AR-1259 CAC-AR-1263	Once/6 months	Each shift
10.	Containment hydrogen	CAC-AR-1259 CAC-AR-1263	Once/6 months	Each shift

Amendment #1

BSEP-1 & 2

TABLE 3.2-15

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Trip Function	Trip Level Setting	Minimum Number of Operable Instrument Channels per Trip System (1)	Action
Reactor High Pressure B21-PS-N045 A,B,C,D	≤ 1120 psig	1	(2)
Reactor Low Water Level #2 B21-LIS-N024A,B B21-LIS-N025A,B	≥ -38 in. indicated level	1	(2)

BSEP-1 & 2

NOTES:

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

BASES:

3.2.B Core Standby Cooling System (CSCS) (Cont'd)

Section 3.5. Whenever an instrument in one subsystem is inoperable the limiting condition for operation as specified in Section 3.5 applies. If an instrument is in more than one subsystem of CSCS, then Section 3.5 is too restrictive and the inoperable channel shall be tripped using special jacks or other permanently installed circuits.

C. Control Rod Blocks

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCHFR does not decrease to 1.0. The trip logic for this function is one out of n; e.g., any trip on one of the six APRMs, eight IRMs, or four SRMs will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. The RBM is an operational guide and aid only and is not needed for rod withdrawal.

The APRM rod block trip is flow referenced and prevents a significant reduction in MCHFR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The rod block trips are set so that MCHFR is maintained greater than 1.0.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCHFR approaches 1.0.

BASES:

3.2.C Control Rod Blocks (Cont'd)

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion; thus, control rod motion is prevented.

When minimum conditions for operation are not met, the required action is to leave the channel in the tripped condition until it is repaired.

D. Radiation Monitoring Systems - Isolation And Initiation Functions

Amendment #1

Two radiation monitors are provided which initiate isolation of the Reactor Building and operation of the standby gas treatment system. The monitors are located in the Reactor Building ventilation duct. Any one upscale trip will initiate the isolation. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation to limit the dose rate at the nearest site boundary to less than the dose rate allowed by 10CFR20.

If the minimum conditions for operation are not met, the Reactor Building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

E. Drywell Leak Detection Monitors

The instrumentation that monitors drywell leak detection provides the information to determine whether Specification 3.6.C. (Coolant Leakage) is met, therefore, the limiting condition for operation is the same as Specification 3.6.C.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3 Reactivity Control	4.3 Reactivity Control
Applicability:	Applicability:
Applies to the operational status of the control rod system.	Applies to the surveillance requirements of the control rod system.
Objective:	Objective:
To assure the ability of the control rod system to control reactivity.	To verify the ability of the control rod system to control reactivity.
Specification:	Specification:
A. Reactivity Limitations	A. Reactivity Limitations
1. Reactivity margin - core loading The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted.	1. Reactivity margin - core loading Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.28 percent Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.
2. Reactivity margin - inoperable control rods a. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met.	2. Reactivity margin - inoperable control rods a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week, when operating above 20% power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above 20% power.

Amendment 1 Amendment 1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.A Reactivity Limitations (Cont'd) c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable. d. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically. e. During reactor power operation, the number of inoperable control rods shall not exceed eight. In addition, during reactor power operation no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). Specification 3.3.A.1 must be met at all times.	4.3.A Reactivity Limitations (Cont'd) b. When it is initially determined that a control rod is incapable of normal insertions, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, the reactor shall be brought to the Cold Shutdown Condition within 24 hours and a shutdown margin test made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted.
B. Control Rods 1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	B. Control Rods 1. The coupling integrity shall be verified for each withdrawn control rod as follows: a. When the rod is withdrawn the first time subsequent to each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to verify instrumentation response.

Amendment 1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B Control Rods (Cont'd)	4.3.B.1 Control Rods (Cont'd)
	b. When the rod is fully withdrawn the first time subsequent to each refueling outage or after maintenance, observe that the drive does not go to the over-travel position.
2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3. a. Control rod patterns shall be established so that the maximum worth of any operable control rod shall be less than 1.43 percent Δk below 30 percent reactor power.	3. a. Prior to control rod withdrawal for startup, the RWM and the RCSC shall be verified as operable. If the RWM is inoperable, a second licensed operator or qualified employee shall be present to verify the following of the rod program.
b. Whenever the reactor is in STARTUP or RUN below 20 percent rated thermal power, the rod sequence control system (RSCS) shall be operable.	b. Prior to the start of control rod withdrawal towards criticality, the capability of the rod sequence control system to properly fulfill its function shall be verified by attempting to select and move a rod in out-of-sequence groups.
c. Whenever the reactor is below 20 percent rated power, the Rod Worth Minimizer shall be operable or a second licensed operator or qualified employee shall verify that the operator at the reactor console is following the control rod program.	c. 1. Proper annunciation of the selection error of at least one out-of-sequence control rod in a fully inserted group shall be verified prior to startup only.
NOTE: The Rod Sequence Control System (RSCS) has been evaluated only through the first refueling outage. A complete re-evaluation is required prior to operations following the first refueling outage.	2. The rod block function of the RWM shall be verified by withdrawing the first rod during startup only as an out-of-sequence control rod no further than to the block point.

Amendment 1

APRIL 1975

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B.3 Control Rods (Cont'd)	4.3.B.3 Control Rods (Cont'd)
d. Whenever the Rod Worth Minimizer and the Rod Sequence Control System are required to be operable, within any sequence (i.e., A or B) no more than one control rod in any 5 x 5 array (at least 4 operable control rods must separate any 2 inoperable ones) may be inoperable. Any rod with a failed "Full in" or "Full out" position switch may be bypassed in the rod sequence control system if actual rod position is known.	d. A second licensed operator or other qualified employee shall verify the conformance to Specification 3.3.B.3.d before a rod may be bypassed in the rod sequence control system.
e. For Unit 2 the rods selected to be used for Select Rod Insert must be consistent with the notch group assignments of either sequence A or B. Mixing of sequence A or B notch groups when selecting rods for the SRI Systems is not allowed.	e. A licensed operator or other qualified employees shall verify the conformance to Specification 3.3.B.3.e prior to withdrawing of control rods or when changing selected rods for Select Rod Insert when at power.
f. If Specifications 3.3.B.3.b through 3.3.B.3.e cannot be met, the reactor shall be shut down.	f. The capability of the RSCS to properly fulfill its function shall be verified by the following tests: Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequence and attempt to move a rod in each. Repeat for all sequences. Group notch portion - For each of the six comparator circuits, go through test initiate; comparator inhibit; verify; reset. On seventh attempt, test is allowed to continue until completion is indicated by illumination of test complete light.

Amendment 1

Amendment 1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B Control Rods (Cont'd)	4.3.B Control Rods (Cont'd)
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During reactor power operation with limiting control rod patterns, as determined by a Plant Engineer, either: a. Both RBM channels shall be operable; or b. Control rod withdrawal shall be blocked; or c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.	5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.
Amendment 1 6. In order to perform the required shutdown margin demonstrations subsequent to any fuel loading operations, to perform tests to verify shutdown margin due to inoperable control rod, or to perform control rod drive scram and/or friction testing and the initial startup test program, the relaxation of the following RSCS restraints is permitted. The sequence restraints imposed on control rod groups A_{12}, A_{34}, B_{12} or B_{34} may be removed for the test period by means of the individual rod position bypass switches.	Amendment 1 6. Prior to control rod withdrawal for startup, verify the conformance to specification 3.3.B.3d before a rod may be bypassed in the RSCS. The requirements to allow use of the individual rod position bypass switches within rod groups A_{12}, A_{34}, B_{12}, or B_{34} of the RSCS during shutdown margin, scram time or friction testing and the initial startup test program are: (a) RWM operable as per specification 3.3.B.3C. (b) After the bypassing of the rods in the RSCS groups A_{12}, A_{34}, B_{12} or B_{34} for test purposes, it shall be demonstrated that movement of the rods in the 50 percent density to the preset power level range is blocked or limited to the single notch mode of withdrawal. (c) A second licensed operator shall verify the conformance to procedures and this Specification.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.C Scram Insertion Times	4.3.C Scram Insertion Times
1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no longer than:	1. After each refueling outage all operable fully withdrawn insequence rods shall be scram time tested during operational hydrostatic testing or during startup from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to synchronizing the main turbine generator initially following restart of the plant. Prior to exceeding 40% of rated power, all untested operable control rods shall be tested as described above.
Above 950 psig % Inserted From Fully Withdrawn — Avg. Scram Insertion Times (sec) 5 — 0.375 20 — 0.90 50 — 2.0 90 — 3.5	
2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no longer than:	2. At 16 week intervals, 10 percent of the control rods capable of movement with control rod drive pressure shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained. If a scram occurs and scram time measurements are available from the scram timing processor, the above 16 week time interval is to start from date of scram. If a scheduled shutdown is planned near the midcycle period, at which time rod scram measurements will be taken for over 50 percent of the operable control rods, the above 16 week interval does not apply.
Above 950 psig % Inserted From Fully Withdrawn — Avg. Scram Insertion Times (sec) 5 — 0.398 20 — 0.954 50 — 2.120 90 — 3.800	

Amendment 1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5 Core and Containment Cooling Systems	4.5 Core and Containment Cooling Systems
Applicability:	Applicability:
Applies to the operational status of the core and containment cooling subsystems.	Applies to the surveillance requirements of the core and containment cooling subsystems which are required when the corresponding limiting condition for operation is in effect.
Objective:	Objective:
To assure the operability of the core and containment cooling subsystems under all conditions for which this cooling capability is an essential response to plant abnormalities.	To verify the operability of the core and containment cooling subsystems under all conditions for which this cooling capability is an essential response to plant abnormalities.
Specification:	Specification:
A. Core Spray and LPCI Subsystems	A. Core Spray and LPCI Subsystems
1. Both core spray subsystems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a cold condition, except as specified in 3.5.A.2 below.	1. Core Spray Subsystem Testing.

Item	Frequency
a. Simulated automatic actuation test	once/operating cycle
b. Pump operability	once/month
c. Motor-operated valve operability	once/month
d. Pump flow rate. Each pump shall deliver at least 4625 gpm against a system head corresponding to a reactor vessel pressure of 113 psig.	once/3 months

Amendment #1

Amendment #1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray and LPCI Subsystem (Cont'd)	4.5.A.1 Core Spray and LPCI Subsystem (Cont'd)
	e. Core spray header Δp instrumentation Check — Once/day Calibrate — Once/3 months Test — Once/3 months
2. From the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.	2. When one core spray subsystem is made or found to be inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable within 24 hours. The operable core spray subsystem shall be demonstrated to be operable daily thereafter until the other core spray subsystem is returned to an operable condition.
3. The LPCI subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a cold condition, except as specified in 3.5.A.4 and 3.5.A.5 below. When the reactor power is greater than 30% of rated power the electrical power to the operator of the LPCI System cross-tie valve shall be removed with the valve in a closed position.	3. LPCI subsystem testing shall be as follows: a. Simulated automatic actuation test — Once/operating cycle b. Pump operability — Once/month c. Motor-operated valve operability except recirculation pump discharge valves and recirculation pump discharge valve bypass valves — Once/month d. Pump flow rate — Once/3 months Each LPCI pump shall deliver 10,300 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray and LPCI Subsystem (Cont'd)	4.5.A Core Spray and LPCI Subsystem (Cont'd)
	Each pair of LPCI pumps discharging to a common header shall deliver 17,000 gpm against a system head corresponding to a reactor vessel pressure of 20 psig. Amendment #1
	e. Recirculation pump discharge valves travel from full open to full closed. — Once/refueling outage
4. From the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days the remaining active components of the LPCI subsystem and all active components of both core spray subsystems and the diesel generators are operable.	4. When one of the RHR (LPCI) pumps is made or found to be inoperable at a time when it is required to be operable, the remaining active components of the LPCI subsystem, the containment cooling subsystem, both core spray subsystems and the diesel generators shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter, until the pump is returned to an operable condition.
5. From the date that the LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystems (including two LPCI pumps) and the diesel generators are operable.	5. When the LPCI subsystem is made or found to be inoperable, both core spray subsystems, the containment cooling subsystem and the diesel generators shall be demonstrated to be operable immediately and daily thereafter until the LPCI subsystem is returned to an operable condition.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.A Core Spray and LPCI Subsystem (Cont'd)	
6. If the requirements of 3.5.A.1 through 5 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.	
3.5.B Containment Cooling Subsystem	4.5.B Containment Cooling Subsystem
1. Both subsystems of containment cooling mode, each including two RHR pumps and two RHR service water pumps, shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor vessel is not vented except as in 3.5.B.2 below.	1. The subsystem of containment cooling mode is tested in conjunction with the test performed on the LPCI systems and given in 4.5.A.3.a, b, and d above. Residual heat removal service water pumps, two pumps operating in parallel will be included in testing, supplying 8000 gpm.
2. Should one RHR pump and/or one RHR service water pump of the components required in 3.5.B.1 above be made or found inoperable, reactor power operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining active components of the suppression pool cooling mode is operable.	2. When one RHR pump and/or one RHR service water pump of the components required in 3.5.B.1 above is made or found to be inoperable, the remaining active components of the containment cooling mode subsystems shall be operable.
3. Should one of the containment cooling subsystems be made or found to be inoperable, reactor power operation is permissible for a period not to exceed seven days provided that during such seven days the other containment cooling subsystem is operable.	3. When one subsystem of the containment cooling mode is made or found to be inoperable when required, the other containment cooling subsystem shall be operable. The operable containment cooling subsystem shall be tested daily thereafter until the inoperable containment cooling components are returned to an operable condition.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.G Engineered Safeguards Compartments Cooling and Ventilation If both unit coolers serving one Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Core Spray or Residual Heat Removal (RHR) pump are out of service, the associated pump shall be considered inoperable for purposes of specifications 3.5.A, 3.5.C, or 3.5.D as applicable.	4.5.G Engineered Safeguards Compartments Cooling and Ventilation The unit coolers for each of the RCIC, HPCI, Core Spray, and RHR pump shall be checked for operability during surveillance testing of the associated pumps.
3.5.H Maintenance of Filled Discharge Pipe Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. When an automatic fill station for the core spray, LPCI, HPCI, or RCIC subsystem is inoperable, the associated discharge pipes will be checked full immediately.	4.5.H Maintenance of Filled Discharge Pipe The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray subsystems, LPCI subsystems, HPCI, and RCIC are filled: 1. Every three months and prior to the testing of the LPCI, HPCI, RCIC, or core spray subsystems, the discharge piping of these systems shall be vented and waterflow observed. 2. Following any period where the LPCI, HPCI, RCIC or core spray subsystems have not been required to be operable, and surveillance has not been performed thus requiring the system to be considered inoperable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service. 3. When an automatic fill station for the core spray, LPCI, HPCI, or RCIC subsystem is inoperable, the associated discharge pipes will be manually filled daily.

Amendment #1

Amendment #1

Amendment #1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6 Primary System Boundary	4.6 Primary System Boundary
Applicability:	Applicability:
Applies to the operating status of the reactor coolant system.	Applies to the periodic examination and testing requirements for the reactor cooling system.
Objective:	Objective:
To assure the integrity and safe operation of the reactor coolant system.	To determine the condition of the reactor coolant system and the operation of the safety devices related to it.
Specification:	Specification:
A. Thermal and Pressurization Limitations	A. Thermal and Pressurization Limitations
1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100 F/hr when averaged over a one-hour period.	1. During heatups and cooldowns, the following temperatures shall be recorded at least every 15 minutes until the difference between any two readings taken over a 45-minute period is less than 5 F. a) Bottom head drain. b) Recirculation loops. c) Bottom head metal temperature in the control rod drive pattern area.
2. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the higher of the temperatures shown on the two curves on Figure 3.6-2 where the dashed line curve, RPV shell beltline region is increased by the expected shift in RT_{NDT} from Figures 3.6-1. During heatup by non-nuclear means, cooldown following nuclear shutdown, low level physics tests, (except at times when the reactor vessel is vented), the reactor vessel shell and fluid temperatures of Specification 4.6.A shall be at or above the	2. Reactor coolant pressure shall be recorded at least every 15 minutes whenever the recirculation loop temperature is below 220 F and the reactor vessel is not vented. A neutron flux dosimeter and material samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM E185-66. The neutron flux dosimeter shall be removed during the first refueling outage and tested to verify or adjust the calculated values of integrated neutron fluence that are

Amendment #1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6 Primary System Boundary (Cont'd)	4.6 Primary System Boundary (Cont'd)
higher of the temperatures of Figure 3.6-3 where the dashed line curve, RPV beltline is increased by the expected shift in RT_{NDT} from Figure 3.6-1. During operation with a critical core, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures of Specification 4.6.A shall be at or above the higher of the temperatures of Figure 3.6-4 where the dashed line curve, RPV beltline, is increased by the expected shift in RT_{NDT} from Figure 3.6-1.	used to determine the shift of RT_{NDT} from Figures 3.6-1. The material surveillance sample frequency for the two units will be performed in accordance with 10CFR50, Appendix H. The estimated time for withdrawal of the first sample will be at one quarter of reactor operating service life, or approximately in 1983.
3. The reactor vessel head bolting studs shall not be under tension unless the temperatures of the vessel head flange and the head is greater than 70 F.	3. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be recorded.
4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50 F of each other.	4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be logged.
5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145 F.	5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and logged.
3.6.B Coolant Chemistry	4.6.B Coolant Chemistry
1. Coolant Activity Limits Whenever the reactor is critical, the limits on iodine activity concentrations in the reactor coolant shall not exceed the equilibrium value of 6.2 μCi/gm of dose equivalent* I-131.	1. During the equilibrium power operation the sampling frequencies of Table 4.6-2 shall apply. Additional samples shall be taken whenever the reactor coolant concentration exceeds .062 μCi/gm of dose equivalent I-131 and one or more of the following conditions are met:

*The following are defined equivalent to 1 μCi of I-131 for this limiting condition: I-132, 29 μCi; I-133, 3.6 μCi; I-134, dose contribution is insignificant, I-135, 12 μCi.

Amendment #1

Amendment #1

LIMITING CONDITONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.B Coolant Chemistry (Cont'd) This limit may be exceeded for a maximum of 48 hours. If the iodine concentration in the coolant exceeds the equilibrium limit by a factor greater than ten, the reactor shall be shutdown, and the steam line isolation valves shall be closed immediately. The reactor shall not be operated under this exception from the equilibrium activity limits for more than three weeks per year.	4.6.B Coolant Chemistry (Cont'd) a. During startup b. Following a power change exceeding 15% rated power in ≤ 1 hr. that results in 4.6.B.1.c being exceeded after correcting proportionally for the power changes c. Following a significant increase** in the equilibrium offgas level at the steam air ejector over a 1 hr. period. The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until two successive samples indicate a decreasing trend below 6.2 μCi/gm dose equivalent I-131. However, at least 3 consecutive samples shall be taken in all cases. A gross or isotopic iodine measurement shall be performed on all samples. If this measurement exceeds 6.2 μCi/gm an isotopic analysis to determine dose equivalent I-131 shall be performed. Samples shall be analyzed within 24 hours after collection.
2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour , except as specified in 3.6.B.3. Conductivity...2 μmho/cm Chloride ion...0.2 ppm	2. During startups and when steaming at rates less than 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

Amendment #1

**The following definition will apply to the term significant increase in offgas level.

a) At release rates less than or equal to 75,000 μCi/sec significant increase means an increase of 10,000 μCi/sec from the previous corresponding power level steady state release rate within 1 hour (at the steam jet air ejector).

b) At release rates greater than 75,000 μCi/sec significant increase means an increase of 15% from the previous corresponding power level steady state release rate within 1 hour.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.B Coolant Chemistry (Cont'd)	4.6.B Coolant Chemistry (Cont'd)
3. For reactor startups and for the first 24 hours after placing the reactor in the power operating condition, the following limits shall not be exceeded: Conductivity...10μ mho/cm Chloride ion...0.2 ppm	3. a. With steaming rates of 100,000 pounds per hour or greater, a reactor coolant sample shall be taken at least every 104 hours and analyzed for conductivity and chloride ion content. b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken daily and analyzed for conductivity and chloride content.
4. Except as specified in 3.6.B.3 above, the reactor coolant water shall not exceed the following limits when operating with steaming rates greater than or equal to 100,000 pounds per hour. Conductivity...2μ mho/cm Chloride...0.5 ppm	
5. If Specification 3.6.B.2 through 4 cannot be met, an orderly shutdown shall be initiated within 24 hours.	
3.6.C. Coolant Leakage	4.6.C Coolant Leakage
1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212 F. reactor coolant leakage into the primary containment from unidentified sources shall not exceed five gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.	1. Reactor coolant system leakage shall be checked by the pump and air sampling system and recorded at least once per day.

LIMITING CONDITONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.E. Jet Pumps	4.6.E. Jet Pumps
1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be placed in a cold shutdown condition within 24 hours.	1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously: a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics. b. The indicated total core flow differs by 10% from the core flow value derived from established power-core flow relationships. c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.
2. Flow indications from each of the 20 jet pumps shall be verified prior to initiation of reactor start-up from a cold shutdown condition.	2. Additionally, when operating with 1 recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.
3. The indicated core flow is the sum of the flow indication from each of the 20 jet pumps. If flow indication failure occurs for 2 or more jet pumps immediate action shall be taken to determine jet pump operability. If flow indication for all but 1 pump cannot be obtained within 12 hours an orderly shutdown shall be in initiated, and the reactor shall be in a cold shutdown condition within 24 hours.	3. The baseline data required to evaluate the conditions in Specifications 4.6.E.1 and 4.6.E.2 will be acquired each operating cycle.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.F. Jet Pump Flow Mismatch	4.6.F. Jet Pump Flow Mismatch
1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.	Recirculation pump speeds shall be checked and logged at least once per day.
2. Following 1-pump operation, the discharge valve of the low-speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed. If Specification 3.6.F.1 cannot be met, 1 recirculation pump shall be tripped.	
3.6.G Structural Integrity	4.6.G Structural Integrity
The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.	The nondestructive inspections listed in Table 4.6-1 shall be performed as specified. The results obtained from compliance with this Specification will be evaluated after five years and the conclusions of this evaluation will be reviewed with the AEC.
H. Condensate Demineralizers	H. Condensate Demineralizers
1. Regeneration of a condensate demineralizing resin charge shall occur before the predicted unused capacity of the resin reaches a minimum value of 30 pounds as chloride ions. Predicted capacity is based on resin salt splitting capacity, integrated flow or flow rate and influent conductivity.	1. The percent of the remaining ion exchanger capacity of the anion resins shall be calculated and logged a. weekly when the influent conductivity is less than 0.3 umho/cm b. daily when the influent conductivity is equal to greater than 0.3 umho/cm.
2. At least one condensate demineralizer influent conductivity instrument shall be operable.	

Amendment #1

TABLE 3.7-1 (Cont'd)

Group	Valve Indentification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
2	Drywell purge exhaust backup valve CAC-V10		1	15	C	SC
2	Containment air purge isolation valve CAC-V15		1	15	C	SC
2	Suppression chamber vent valve CAC-V22		1	15	C	SC
2	Drywell purge exhaust backup valve bypass valve CAC-V23		1	15	C	SC
2	Suppression chamber makeup and containment atmosphere dilution inlet valve CAC-V47		1	15	O	GC
2	Drywell makeup and containment atmosphere dilution inlet valve CAC-V48		1	15	O	GC
2	Drywell vent isolation valve CAC-49		1	15	C	SC
2	Drywell vent backup valve CAC-50		1	15	C	SC
2	Containment atmosphere dilution inlet valve CAC-V55		1	15	C	SC
2	Containment atmosphere dilution inlet bypass valve CAC-V56		1	15	C	SC

BSEP-1 & 2

TABLE 3.7-1 (Cont'd)

Group	Valve Indentification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
2	RHR shutdown cooling supply isolation valves E11-F008 E11-F009	1	1	30	C	SC
2	RHR injection isolation valves E11-F015A, B		2	30	C	SC
2	Reactor vessel head spray isolation valves E11-F022 E11-F023	1	1	30	C	SC
2	RHR discharge isolation valves to radwaste E11-F040 E11-F049		2	30	C	SC
2	RHR process sampling valves E11-F079A, B E11-F080A, B	2	2	5	C	SC
2	Drywell equipment drain discharge isolation valves G16-F019 G16-F020	1	1	20	O	GC
2	Drywell floor drain discharge isolation valves G16-F003 G16-F004	1	1	20	O	GC
3	Reactor water cleanup system isolation valves G31-F001 G31-F004	1	1	35	O	GC

BASES:

3.7.B & 3.7.C Continued

efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Amendment #1

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the standby gas treatment system is not required.

BASES:

4.7.B and 4.7.C Standby Gas Treatment System and Secondary Containment

Initiating Reactor Building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containmnnt provides an adequate test of the operation of the Reactor Building isolation valves, leak tightness of the Reactor Building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Periodic testing gives sufficient confidence of Reactor Building integrity and standby gas treatment system performance capability.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1-1972. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

This committee, as shown on Figure 6.5-2 includes as members the Director of Technical Services as Chairman, the Manager of Nuclear Generation, the Manager of Licensing and Technological Services, the Vice President - Power Plant Engineering, the Vice President - Bulk Power Supply, the Manager of Nuclear Engineering, Manager of Fuel, the Principal Radiation Control Engineer, a member of the supervisory staff of each plant, and a non-Company member. The Director of Research serves as Vice Chairman. A Technical Services Senior Engineer serves as Secretary.

6.5.2.3 Alternates

Alternate members shall be appointed in writing by the CNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNSC activities at any one time.

6.5.2.4 Qualifications

Each Company Nuclear Safety Committee member and alternate shall have at least a Bachelor's Degree in Engineering or the Physical Sciences or equivalent and a minimum of five years of professional level experience in the technical discipline represented.

6.5.2.5 Additional Qualifications

In addition of 6.5.2.4 above, at least four members and alternates shall have a minimum of three years experience in nuclear services, nuclear plant operation or nuclear engineering.

Amendment #1

6.5.2.6 Consultants

Consultants shall be utilized as determined by the Chairman to provide expert advice to the CNSC.

6.5.2.7 Meeting Frequency

The Committee shall meet at least once each calendar quarter during the initial year of operation following the initial fuel loading; thereafter, at least twice per year.

6.5.2.8 Quorum

A quorum of the Committee consists of the Chairman or Vice Chairman plus five members present except that no more than a minority of the quorum shall be members of the supervisory staff of the nuclear plants.

Amendment #1

6.5.2.9 Review

The Company Nuclear Safety Committee shall review;

a) The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c) Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d) Proposed changes in Technical Specifications or licenses.

e) Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g) ABNORMAL OCCURRENCES, as defined in Section 1.0 of these Technical Specifications.

h) Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i) Reports and meeting minutes of the Plant Nuclear Safety Committee.

Amendment #1

6.5.2.10 Audits

Semi-annual audits shall be conducted in accordance with written procedures under the cognizance of the CNSC The areas of audit shall include:

a) Compliance with internal rules and procedures, Technical Specifications, and applicable license requirements.
b) Implementation of operating requirements, training, and qualifications of plant operating staff.
c) The results of corrective actions taken following abnormal occurrences.
d) Any other area of plant operation considered appropriate by the CNSC or the Vice President - Bulk Power Supply Department.
e) The Plant Emergency Plan and implementing procedures.

6.5.2.11 Authority

The Company Nuclear Safety Committee shall function as an advisory body. The Committee shall have access to the operating record files and operating personnel to perform the audit function and may request such written or oral reports and/or analyses deemed necessary to carry out the committee's responsibilities.

6.5.2.12 Records

Records of activities shall be prepared, approved and distributed as indicated below:

a) Minutes of each CNSC meeting shall be prepared, approved and forwarded to the Vice President-Bulk Power Supply Department within 14 days following each meeting.
b) Reports of reviews encompassed by Section 6.5.2.9 e, f, g and h above, shall be prepared, approved and forwarded to the Vice President-Bulk Power Supply Department within 14 days following completion of the review.
c) Audit reports encompassed by Section 6.5.2.10 above, shall be forwarded to the Vice President-Bulk Power Supply Department and to the management positions responsible for the areas audited within 30 days after completion of the audit.

Amendment #1

6.5.2.13 Procedures

Written administrative procedures for committee operation shall be prepared and maintained.

Table 3.5-4

GASEOUS WASTE SYSTEM
LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLERS

Process Stream or Release Point	Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	NG	Measurement I	Measurement Part	Measurement H-3
Condenser/Air Ejector (before gas treatment system)	X	X	X	X	X			
Augmented Off Gas System Process Monitor (after gas treatment system)	X	X	X	X	X	X		
Main Stack	X		X	X	X	X	X	X
Building Ventilation Systems								
Reactor Building	X	X	X	X	X	X	X	X
*Radwaste Building	X		X	X	X	X	X	X
Turbine Building	X		X	X	X	X	X	X
**Fuel Handling & Storage Building	X		X	X	X	X	X	X
***Mechanical Vacuum Pump	X		X	X	X	X	X	X
***Turbine Gland Seal Condenser	X		X	X	X	X	X	X
****Decontamination Room			X				X	

*Radwaste Building ventilation is routed to the main stack where continuous monitoring is provided.

**Fuel handling and storage area is vented through the Reactor Building ventilation system.

***The off gas from the mechanical vacuum pump is discharged downstream of the turbine gland seal condenser vent which discharges to the main stack where continuous monitoring is provided.

****Continuous monitoring whenever the decontamination room exhaust fan is operating.

Amendment #1

Amendment #1

Table 3.5-5

AVERAGE ENERGY PER DISINTEGRATION

Isotope	$\bar{E}_{\gamma_i}$ Mev/dis	(Ref)	$\bar{E}_{\beta i}$ Mev/dis[3]	(Ref)
Kr-83m	0.00248	(1)	0.0371	(1)
Kr-85	0.0022	(1)	0.250	(1)
Kr-85m	0.159	(1)	0.253	(1)
Kr-87	0.793	(1)	1.32	(1)
Kr-88	1.95	(1)	0.377	(1)
Kr-89	2.22	(2)	1.37	(2)
Kr-90	2.10	(2)	1.01	(2)
Xe-131m	0.0201	(1)	0.143	(1)
Xe-133	0.0454	(1)	0.135	(1)
Xe-133m	0.042	(1)	0.19	(1)
Xe-135	0.247	(1)	0.317	(1)
Xe-135m	0.432	(1)	0.095	(1)
Xe-137	0.194	(1)	1.64	(1)
Xe-138	1.18	(1)	0.611	(1)

(1) ORNL-4923, Radioactive Atoms - Supplement I, M.S. Martin, November 1973.
(2) NEDO-12037, "Summary of Gamma and Beta Emitters and Intensity Data; M. E. Meek, R. S. Gilbert, January 1970. (The average β energy was computed from the maximum energy using the ICRP II equation, not the 1/3 value assumption used in this reference.)
(3) The average β energy includes conversion electrons.