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See draft Tech Specs

SUBJECT: Forwards comments on final draft Tech Specs categorized as either error or improvement item. Error items denote changes which should be reflecte in plant design. Marked-up Tech Specs also encl.

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CP&L

Carolina Power & Light Company

SERIAL: NLS-86-273

JUL 30 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NO. 1 - DOCKET NO. 50-400
COMMENTS ON FINAL DRAFT TECHNICAL SPECIFICATIONS

Dear Mr. Denton:

Carolina Power & Light Company (CP&L) submits comments on the Final Draft Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant (SHNPP). Attachment 1 provides 72 comments (record numbers 700 through 778 with the exception of record numbers 709, 711-715, and 717) on the TS as well as a justification for each comment. Attachment 2 provides a marked-up copy of the TS pages for each of the comments provided in Attachment 1.

The comments are categorized as either an error or improvement item. The term error is used to denote changes required to have the TS reflect the plant design. Comments categorized as an error must be incorporated into the SHNPP TS to allow CP&L to certify the accuracy of the TS. The term improvement item is used to denote those significant changes that would improve the quality of the SHNPP TS and other insignificant typographical errors.

CP&L will certify that the SHNPP TS are consistent with the Final Safety Analysis Report and the as-built plant upon satisfactory resolution of Final Draft comments.

If you have any questions, please contact Mr. Gregg A. Sinders at (919) 836-8168.

Yours very truly,

S. R. Zimmerman

S. R. Zimmerman
Manager
Nuclear Licensing Section

SRZ/GAS/vaw (4022GAS)

Attachments

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Dr. J. Nelson Grace (NRC-RII)
Mr. R. A. Benedict (NRC)
Wake County Public Library

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Attachment 1 to NLS-86-273

Final Draft TS Comments

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 700 Comment Type: ERROR

LCO Number: 3.03.03.11 Page Number: 3/4 3-91

Section Number: TABLE 4.3-9

Comment:

ITEM 4.a.2 - CHANGE THE DIGITAL CHANNEL OPERATIONAL TEST FREQUENCY FROM "Q(1)" TO "Q(2)".

Basis

THIS CHANGE IS NECESSITATAED DUE TO THE FACT THAT THE WRGM DOES NOT HAVE AN ALARM/TRIP FUNCTION AS DESCRIBED IN NOTE 1. RATHER, THE WRGM HAS ONLY THE ALARM FUNCTION AS DESCRIBED IN NOTE 2.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 701 Comment Type: ERROR

LCO Number: 3.07.06 Page Number: 3/4 7-15,16 & B

Section Number: VARIOUS

Comment:

ITEMS 4.7.6.b.1, 4.7.6.d.4, 4.7.6.e, 4.7.6.f AND BASES 4.7.6 - CHANGE ANSI N510-1975 TO ANSI N510-1980 IN ALL PLACES

Basis

THIS CHANGE IS MADE FOR CONSISTENCY WITH THE FSAR.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 702 | Comment Type: ERROR

LCO Number: 3.07.07 | Page Number: 3/4 7-17, 18 &B

Section Number: VARIOUS

Comment:

ITEMS 4.7.7.b.1, 4.7.7.d.5, 4.7.7.e, 4.7.7.f AND BASES 4.7.7 - CHANGE ANSI N510-1975 TO ANSI N510-1980.

Basis

THIS CHANGE IS MADE FOR CONSISTENCY WITH THE FSAR.

CP&L Comments

SNPP Proof and Review Technical Specifications

Record Number: 703 Comment Type: ERROR

LCO Number: 4.09.12 Page Number: 3/4 9-14,15,16

Section Number: VARIOUS

Comment:

ITEMS 4.9.12.b.1, 4.9.12.d.5, 4.9.12.e, 4.9.12.f AND BASES - CHANGE ANSI N510-1975 TO ANSI N510-1980.

Basis

THIS CHANGE IS NECESSARY FOR CONSISTENCY WITH THE FSAR.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 704 Comment Type: IMPROVEMENT

LCO Number: 3.06.02.01 Page Number: 3/4 6-11

Section Number: 4.6.2.1.c.3

Comment:

REWORD ITEM 3 TO THE FOLLOWING:
Verifying that coincident with an indication of containment spray pump running, each automatic valve from the sump and RWST actuates to its appropriate position following an RWST Lo-Lo test signal.

Basis

THIS REWORDING IS NECESSARY FOR TWO REASONS. THE FIRST REASON IS TO ENSURE THAT THE CLOSURE OF THE RWST SUCTION VALVE ON SWITCHOVER IS TESTED. THE SECOND IS THAT PER FSAR FIGURE 7.2.1-3, THE SWITCHOVER REQUIRES THE PRESENCE OF A PUMP RUNNING SIGNAL NOT AN SI SIGNAL AS SUCH.

Record Number: 705 Comment Type: ERROR

LCO Number: 3.03.03.06 Page Number: 3/4 3-66

Section Number: FOOTNOTE *

Comment:

IN THE FOOTNOTE, CHANGE THE WORD "TAILPIPE" TO "SAFETY VALVE PIPING".

Basis

THIS CHANGE IS REQUIRED DUE TO THE PHYSICAL LOCATION OF THE TEMPERATURE SENSOR. THE SENSOR IS IN FRONT OF THE VALVE, NOT AFTER AS THE TERM TAILPIPE WOULD IMPLY.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 706 Comment Type: ERROR

LCO Number: 3.08.04.02 Page Number: 3/4 8-41,42

Section Number: TABLE 3.8-2

Comment:

THE LAST SEVEN ITEMS ON PAGE 3/4 8-41 AND THE FIRST ITEM ON PAGE 3/4 8-42 - CHANGE THE BYPASS DEVICE COLUMN FROM "YES" TO "NO*".

Basis

THIS CHANGE IS REQUIRED DUE TO RECENT PLANT MODIFICATIONS. THE RESULT OF THESE MODIFICATIONS IS THAT THE THERMAL OVERLOAD BYPASS FUNCTION IS NOW COVERED BY INHERENT FEATURES DESIGNED INTO THE CIRCUITRY AND THERE IS NO LONGER A "BYPASS DEVICE" TO BE TESTED.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 707 Comment Type: ERROR

LCO Number: B 3/4.04.05 Page Number: B 3/4 4-3

Section Number: B 3/4.4.5

Comment:

IN THE LAST PARAGRAPH OF THE SECTION, CHANGE "SPECIFICATION 6.9.2" TO "SPECIFICATION 4.4.5.5.c".

Basis

THIS CHANGE IS TO PROVIDE CONSISTENCY WITH THE BODY OF THE SPECIFICATIONS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 708 Comment Type: ERROR

LCO Number: 3.08.04.01 Page Number: 3/4 8-21 ETC.

Section Number: TABLE 3.8-1

Comment:

CHANGE THE TABLE PER THE ATTACHED MARKED UP PAGES.

Basis

THIS CHANGE IS TO CORRECT TYPOGRAPHICAL ERRORS IN THE TABLE WHICH HAVE EXISTED FROM THE FIRST CP&L SUBMITTAL. ALSO, THE TABLE HAS BEEN REVISED TO INCLUDE ADDITIONAL DEVICES AS DETERMINED BY THE A/E.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 710 | Comment Type: ERROR

LCO Number: 3.06.03 | Page Number: 3/4 6-16 ETC.

Section Number: TABLE 3.6-1

Comment:

CHANGE THE TABLE PER THE ATTACHED MARKED UP PAGES.

Basis

THIS CHANGE IS TO CORRECT TYPOGRAPHICAL ERRORS IN THE INFORMATION PREVIOUSLY PROVIDED BY CP&L.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 716 Comment Type: ERROR

LCO Number: 3.03.02 Page Number: 3/4 3-39

Section Number: TABLE 3.3-5

Comment:

CHANGE THE RESPONSE TIME OF ITEM 12 FROM "41 sec"
TO "< or = 41 sec"

Basis

THIS IS A TYPOGRAPHICAL ERROR NOT CORRECTED IN
PREVIOUS SUBMITTALS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 718 Comment Type: IMPROVEMENT

LCO Number: 3.05.01 Page Number: 3/4 5-1

Section Number: 3.5.1.a

Comment:

ADD TO THE END OF 3.5.1a "with power supply circuit breaker open".

Basis

THIS CHANGE IS PROPOSED TO PROVIDE GREATER CLARITY AND CONSISTENCY BETWEEN THE LCO AND THE SURVEILLANCE.

CP&L Comments

HNPP Proof and Review Technical Specifications

Record Number: 719 Comment Type: ERROR

LCO Number: B 3/4.01.02 Page Number: B 3/4 1-2

Section Number: B 3/4.1.2

Comment:

THE FIRST LINE IN PARAGRAPHS 2 AND 3 - CHANGE "200 F" TO "350 F".

Basis

THE CHANGE IS NEEDED FOR CONSISTENCY WITH LCO's 3.1.2.1 AND 3.1.2.2 FOR CSIP OPERABILITY. THE TEMPERATURES ON B 3/4 1-3 DO NOT NEED TO CHANGE BASED ON BORATED WATER SOURCE AVAILABILITY IN LCO's 3.1.2.5 AND 3.1.2.6. THIS IS THE SAME AS THE BYRON BASES.

CP&L Comments

NPP Proof and Review Technical Specifications

Record Number: 720 — Comment Type: IMPROVEMENT

LCO Number: B 3/4.01.02 — Page Number: B 3/4 1-3

Section Number: B 3/4.1.2

Comment:

> ADD TO THE END OF THE NEXT TO LAST PARAGRAPH OF SECTION B 3/4.1.2 THE FOLLOWING SENTENCE:
> The RWST temperature was selected to be consistent with analytical assumptions for containment heat load.

Basis

> THIS CHANGE IS TO PROVIDE ADDITIONAL INFORMATION FOR THE TECH SPEC USERS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 721 Comment Type: IMPROVEMENT

LCO Number: B 3/4.06.01.04 Page Number: B 3/4 6-1

Section Number: B 3/4.6.1.4

Comment:

ADD TO THE END OF THE SECOND PARAGRAPH THE FOLLOWING SENTENCE:
The -1" wg was chosen to be consistent with the initial assumptions of accident analyses.

Basis

THIS CHANGE IS TO PROVIDE ADDITIONAL INFORMATION FOR TECH SPEC USERS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 722 Comment Type: IMPROVEMENT

LCO Number: 3.07.01.02 Page Number: 3/4 7-4

Section Number: 4.7.1.2.1.a

Comment:

ITEM 4.7.1.2.1.a.1 - CHANGE "1510" TO "1590".
ITEM 4.7.1.2.1.a.2 - CHANGE "1450" TO "1510".

Basis

NEW VALUES HAVE BEEN PROVIDED BY THE A.E. FOR THESE DISCHARGE PRESSURES BASED ON PUMP CURVES AND TESTING RESULTS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 723 | Comment Type: ERROR

LCO Number: 3.04.06.02 | Page Number: 3/4 4-24

Section Number: 4.4.6.2.1.b

Comment:

CHANGE THE WORD "DISCHARGE" TO "FLOW MONITORING SYSTEM".

Basis

THIS CHANGE PROVIDES CONSISTENCY WITH THE NOMENCLATURE OF 3.4.6.1.b AND 4.4.6.1.b AND IS A MORE ACCURATE REPRESENTATION OF THE SYSTEM.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 724 Comment Type: IMPROVEMENT

LCO Number: 3.08.01.01 & .02 Page Number: 3/4 8-1 & 11

Section Number: 3.8.1.1bl & 2bl

Comment:

ITEMS 3.8.1.1.bl AND 3.8.1.2.bl - CHANGE "92.5%" TO "85%".

Basis

A RESCALING OF THE TANK INSTRUMENTATION HAS CHANGED THE PERCENT INDICATED LEVEL FOR THE REQUIRED GALLONAGE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 725 Comment Type: ERROR

LCO Number: 3.03.02 Page Number: 3/4 3-18 & 22

Section Number: TABLE 3.3-3

Comment:

ITEMS 1.b AND 4.b - IN THE APPLICABLE MODE COLUMN, ADD MODE 4 TO BOTH ITEMS.

Basis

RECENT CHANGES REQUESTED BY THE NRC STAFF ON ITEMS 1c AND 4e ADDED MODE 4. THIS CHANGE IS NECESSARY TO ENSURE ACTUATION RELAYS ARE ALSO OPERABLE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 726 | Comment Type: IMPROVEMENT

LCO Number: 3.03.02 | Page Number: 3/4 3-23 AND 27

Section Number: TABLE 3.3-3

Comment:

ITEM 6a - CHANGE THE ACTION TO "23"
TABLE ACTION 23 - REWORD THE NOTE TO THE FOLLOWING:

With the number of OPERABLE channels one less than the Total Number of Channels, declare the affected component inoperable and take the appropriate ACTION per the specific component LCO statement.

Basis

ITEMS 6a AND 4al IN THIS TABLE WERE THE ONLY USES OF ACTION 23 AND BOTH WERE IN SUBSTANTIAL DISAGREEMENT WITH THEIR RESPECTIVE COMPONENT LCO's (3.7.1.2 AND 3.7.1.5). THIS CHANGE ENSURE CONSISTENCY BETWEEN PORTIONS OF THE SPECIFICATIONS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 727

Comment Type: ERROR

LCO Number: 3.03.03.05.a

Page Number: 3/4 3-64

Section Number: TABLE 3.3-9

Comment:

CHANGE TABLE 3.3-9 TO THE ATTACHED MARKUP.

Basis

THIS CHANGE IS NECESSARY FOR THE TECH SPECS TO ACCURATELY REFLECT THE SHNPP DESIGN.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 728 | Comment Type: ERROR

LCO Number: 3.05.01 | Page Number: 3/4 5-1

Section Number: FOOTNOTE *

Comment:

CHANGE THE WORD "PRESSURIZER" TO "RCS".

Basis

PRESSURIZER PRESSURE, AS SUCH, IS NOT READ AT SHNPP BELOW 1700 psig. RATHER, IT IS THE RCS PRESSURE WHICH IS READ IN THIS RANGE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 729 Comment Type: IMPROVEMENT

LCO Number: 3.05.02 Page Number: 3/4 5-6

Section Number: 4.5.2.g

Comment:

DELETE THE COLUMN FOR HPSI SYSTEM EBASCO VALVE NO. (FSAR).

Basis

THE FSAR HAS BEEN REVISED IN AMENDMENT 27 SO THE REVIEW AID OF THE ADDITIONAL COLUMN IS NO LONGER NECESSARY.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 730 Comment Type: ERROR

LCO Number: 3.06.01.06 Page Number: 3/4 6-8

Section Number: 3.6.1.6

Comment:

CHANGE "SPECIFICATION 4.6.1.6" TO "SPECIFICATION 4.6.1.6.1".

Basis

TYPO

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 731

Comment Type: IMPROVEMENT

LCO Number: 3.07.06

Page Number: 3/4 7-15

Section Number: 4.7.6.d.1

Comment:

CHANGE THE FIRST LINE OF SURVEILLANCE TO THE FOLLOWING:
"Verifying that, either a safety injection or a high radiation test signal, the system....

Basis

THIS CHANGE IS TO CLARIFY THAT TWO DIFFERENT TESTS ARE INVOLVED IN MEETING THIS SURVEILLANCE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 732 | Comment Type: IMPROVEMENT

LCO Number: 3.08.01.01 | Page Number: 3/4 8-1,2 & 3

Section Number: ACTIONS

Comment:

CHANGE FOOTNOTE # TO THE FOLLOWING:

Activities which normally support testing pursuant to 4.8.1.1.2a.4 which would render the diesel inoperable (e.g. air roll) shall not be performed for testing required by this ACTION statement.

Basis

THE CURRENT WORDING HAS PROVEN TO BE CONFUSING TO PLANT PERSONNEL. THIS WORDING IS CLEARER BUT DOES NOT CHANGE THE INTENT OF THE ORIGINAL FOOTNOTE IN ANY WAY.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 733 Comment Type: IMPROVEMENT

LCO Number: 3.08.01.01 Page Number: 3/4 8-7

Section Number: 4.8.1.1.2.f.6.c

Comment:

IN THE LAST LINE OF THE SURVEILLANCE, CHANGE THE WORD "OR" TO "IN CONJUNCTION WITH".

Basis

AS WRITTEN, THIS TEST IS NOT CONSISTENT WITH THE SUBSECTION IN WHICH IT IS WRITTEN. THE PROPOSED WORDING CHANGE IS CONSISTENT WITH THE SUBSECTION AND IS THE SAME AS PROVIDED IN DRAFT REV. 5, CALLAWAY AND MILLSTONE TECH SPECS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 734 Comment Type: IMPROVEMENT

LCO Number: 3.09.01 Page Number: 3/4 9-2

Section Number: TABLE 4.9-1

Comment:

REVISE TABLE PER THE ATTACHED MARKUP.

Basis

THESE CHANGES ARE PROPOSED FOR CONSISTENCY WITHIN THE TABLE AND TO PROVIDE ADDITIONAL INFORMATION USEFUL TO PLANT PERSONNEL.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 735 Comment Type: ERROR

LCO Number: 3.09.12 Page Number: 3/4 9-15

Section Number: 4.9.12.d.2

Comment:

DELETE "(UNLESS ALREADY OPERATING)".

Basis

IN ORDER TO PROPERLY CONDUCT THIS TEST, THE FAN MUST BE STOPPED PRIOR TO THE START OF THE TEST. SHNPP FANS DO NOT REDIRECT FLOW.THEREFORE,IF THE FAN IS ALREADY OPERATING, NO CONCLUSION COULD BE REACHED REGARDING A SATISFACTORY COMPLETION OF THE TEST.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 736 Comment Type: ERROR

LCO Number: B 3/4.08.01.01 Page Number: B 3/4 8-2

Section Number: B 3/4.8.1.1

Comment:

IN THE SECOND PARAGRAPH OF THE PAGE, CHANGE "IN ACCORDANCE WITH" TO "BASED UPON".

Basis

THE LATEST NRC STAFF GUIDANCE WAS PROVIDED FOR THE SHNPP DIESEL SPECIFICATION. THIS GUIDANCE DIFFERS IN SOME DETAILS FROM THAT PROVIDED IN REG GUIDE 1.108.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 737 Comment Type: ERROR

LCO Number: 5.07.01 Page Number: 5-8

Section Number: TABLE 5.7-1

Comment:

IN THE DESIGN CYCLE OR TRANSIENT COLUMN FOR THE REACTOR COOLANT SYSTEM 10 AUXILIARY SPRAY ACTUATION CYCLES, CHANGE ",625 F" TO "Greater than 320 F but less than 625 F."

Basis

THIS CHANGE IS REQUIRED TO MAKE THE SPECS MORE ACCURATE. THE CYCLE IS FOR THE TEMPERATURE RANGE <320 F TO >650 F. ACTUATION BELOW 320 F DOES NOT APPLY TO THIS CYCLIC LIMIT.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 738 | Comment Type: IMPROVEMENT

LCO Number: 3.06.01.03 | Page Number: 3/4 6-4

Section Number: ACTION a.1

Comment:

IN THE FIRST LINE OF THE ACTION, PLACE A "*" AFTER THE WORD "CLOSED".

ADD A NEW FOOTNOTE TO THE BOTTOM OF THE PAGE AS FOLLOWS:

* Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

Basis

THIS FOOTNOTE IS NECESSARY TO PERMIT REPAIR AND POST-MAINTENANCE TESTING OF AN INNER AIR LOCK DOOR. THE CUMULATIVE TIME IS SHORT ENOUGH TO PRECLUDE ADVERSE CONSEQUENCES.
THIS FOOTNOTE IS CONSISTENT WITH THE AIRLOCK TECH SPECS RECENTLY APPROVED FOR THE MILLSTONE STATION.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 739

Comment Type: ERROR

LCO Number: 3.08.01.01

Page Number: 3/4 8-7

Section Number: 4.8.1.1.2.f.10

Comment:

ADD A NEW ITEM 4.8.1.1.2.f.10.f AS FOLLOWS:
f. Loss of generator potential transformer circuit.

Basis

THE ADDITION OF THIS ITEM IS NECESSARY TO ENSURE THAT ALL DIESEL LOCKOUT FEATURES ARE TESTED.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 740 Comment Type: ERROR

LCO Number: B 3/4.08.01 Page Number: B 3/4 8-1

Section Number: B 3/4 8.1

Comment:

IN THE LAST PARAGRAPH OF THE PAGE, DELETE THE PHRASE "AND THAT THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP IS OEPRABLE."

Basis

THIS CHANGE IS REQUIRED FOR CONSISTENCY WITH THE ACTION STATEMENT OF 3.8.1.1. AS DIRECTED BY MR. J.T. BEARD OF THE NRC, THE REQUIREMENT THAT THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP BE OPERABLE WAS CHANGED TO PROVIDE DIRECTION ONLY IF ALL THREE FEEDWATER PUMPS ARE INOPERABLE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 741 Comment Type: IMPROVEMENT

LCO Number: B 3/4.08.04 Page Number: B 3/4 8-3

Section Number: B 3/4 .8.4

Comment:

IN THE SECOND PARAGRAPH, DELETE ALL REFERENCES TO FUSES PER THE ATTACHED MARKUP.

Basis

THIS CHANGE IS NECESSARY DUE TO THE CHANGE PREVIOUSLY APPROVED BY THE NRC WHICH DELETED SURVEILLANCE TESTING OF FUSES. WHEN THE CHANGE WAS MADE TO THE SURVEILLANCES, THE BASES CHANGES WERE INADVERTANTLY MISSED.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 742 Comment Type: ERROR

LCO Number: 3.03.02 Page Number: 3/4 3-38

Section Number: TABLE 3.3-5

Comment:

ITEMS 4b, 6 AND 7 - CHANGE THE RESPONE TIME FROM "7" TO "12" .

Basis

THIS CHANGE IS REQUIRED DUE TO THE VALVE CLOSURE TIME OF THE MSIV BYPASS LINE. THE VALVE CLOSURE TIME OF THE BYPASS LINE IS 10 sec. THIS TIME ADDED TO THE 2 sec SIGNAL PROCESSING TIME YEILDS A MSIV ISOLATION TIME OF 12 sec. THIS IS CONSISTENT WITH THE TIMES SHOWN IN THE FSAR AND ON TABLE 3.6-1 OF THE FINAL DRAFT TECH SPECS.

CP&L Comments

NPP Proof and Review Technical Specifications

Record Number: 743 Comment Type: ERROR

LCO Number: 6.05.03.01 Page Number: 6-11

Section Number: 6.5.3.1

Comment:

CHANGE THE LAST SENTENCE OF THE PARAGRAPH TO READ AS FOLLOWS:
They shall also evaluate all CP&L LER's for their potential applicability to other CP&L units.

Basis

SEE ITEM 744
THIS CHANGE IS NEEDED TO ACCURATELY REFLECT THE EXACT ORGANIZATION THAT PERFORMS THE VARIOUS REVIEWS. ALL ITEMS MENTIONED IN THE FINAL DRAFT ARE STILL COVERED, BUT HAVE BEEN MOVED TO THEIR PROPER PLACE.

CP&L Comments

INPP Proof and Review Technical Specifications

Record Number: 744 Comment Type: ERROR

LCO Number: 6.02.03.01 Page Number: 6-6

Section Number: 6.2.3.1

Comment:

INSERT IN THE SECOND LINE AFTER "industry advisories" THE FOLLOWING WORDING "(including information forwarded from INPO from their evaluation of all industry LER's).

Basis

SEE ITEM 743
THIS CHANGE IS NEEDED TO ACCURATELY REFLECT THE EXACT ORGANIZATION THAT PERFORMS THE VARIOUS REVIEWS. ALL ITEMS MENTIONED IN THE FINAL DRAFT ARE STILL COVERED, BUT HAVE BEEN MOVED TO THEIR PROPER PLACE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 745 Comment Type: IMPROVEMENT

LCO Number: 6.05.03.09 Page Number: 6-13

Section Number: 6.5.3.9.e

Comment:

IN THE SECOND LINE DELETE THE WORD "AND".
REWORD THE LAST LINE TO THE FOLLOWING:
...plant safety-related structures, systems, or components which require written notification to the commission.

Basis

THE DELETION OF THE WORD "AND" IS A GRAMMATICAL CORRECTION. THE ADDITION OF THE WORDS "SAFETY-RELATED" IS TO PROVIDE GREATER SPECIFICITY TO THE REQUIREMENT. AND, THE CHANGE TO THE END OF THE SENTENCE IS FOR CONSISTENCY WITH THE WORDING OF ANSI N18.7 AND WITH THE WORDING OF BOTH THE ROBINSON AND BRUNSWICK TECH SPECS.
CP&L HAS A CORPORATE PROGRAM IN THIS AREA AND IT IS NECESSARY THAT THERE BE CONSISTENCY BETWEEN THE REQUIREMENTS FOR THE VARIOUS PLANTS. THIS CHANGE PROVIDES THAT INTERNAL CONSISTENCY AS WELL AS BEING IN CONFORMANCE TO THE APPLICABLE STANDARD.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 746 Comment Type: IMPROVEMENT

LCO Number: 6.05.03.09 Page Number: 6-13

Section Number: 6.5.3.9.i

Comment:

DELETE ITEM i AND RELETTER j TO i. ALSO IN ITEM 6.5.3.10 CHANGE ITEM j TO i.

Basis

CP&L HAS A CORPORATE PROGRAM FOR 3 REACTOR SITES AND IT IS HIGHLY DESIRABLE TO KEEP THE REQUIREMENTS FOR ALL UNITS COMPATABLE. THIS ITEM DOES NOT APPEAR IN THE BRUNSWICK NOR ROBINSON SPECIFICATIONS. IN ADDITION, THE REQUIREMENT IS SO BROAD AND VAQUELY WORDED THAT IT APPEARS ALMOST IMPOSSIBLE TO SET UP AN AUDITABLE PROGRAM TO COVER THE ITEM. IT APPEARS, IN GENERAL, TO COVER THE SAME GROUND AS 10CFR21 AND AS SUCH IS ALREADY COVERED BY ITEM e ABOVE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 747

Comment Type: IMPROVEMENT

LCO Number: B 3/4.01.02

Page Number: B 3/4 1-2 & 3

Section Number: B 3/4.1.2

Comment:

IN THE SECOND PARAGRAPH ON PAGE B 3/4 1-2 AND IN THE SECOND FULL PARAGRAPH ON PAGE B 3/4 1-3 CHANGE "2000 ppm" TO "2000-2200 ppm".

Basis

THIS CHANGE IS REQUIRED FOR CONSISTENCY BETWEEN THE BASES AND THE SPECIFICATIONS OF SECTION 3.1.2.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 748 Comment Type: IMPROVEMENT

LCO Number: 3.06.02.02 Page Number: 3/4 6-12

Section Number: 4.6.2.2.d

Comment:

CHANGE "17 gpm" TO "20 PLUS OR MINUS 0.5 ppm". AND CHANGE "100,000 gal" TO "436,000 gal".

Basis

THE CHANGES IN THE VALUES TO THE FLOW RATES ARE CONSISTENT WITH THE FULL RWST. THIS PLACES THE TEST AT A BETTER POINT ON THE CURVE SINCE IT WILL SHOW THAT THE HIGH EDUCTOR FLOW RATE LIMIT IS MET. THESE NEW VALUES ARE CONSISTENT WITH THE PREVIOUS VALUES, BUT THIS IS FELT TO BE A BETTER TEST.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 749 Comment Type: ERROR

LCO Number: 3.08.03.01 Page Number: 3/4 8-17

Section Number: ACTION C

Comment:

CHANGE ACTION C TO READ AS FOLLOWS:

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, reenergize the 118-volt A.C. bus through its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Basis

THIS CHANGE IS NECESSARY TO CLARIFY THAT AN INVERTER MUST BE RECONNECTED TO THE VITAL BUS WITHIN 24 HOURS. AS CURRENTLY WRITTEN, THE REQUIREMENT IS ONLY TO RECONNECT THE INVERTER TO THE D.C. POWER SUPPLY BUT NOT TO RECONNECT THE INVERTER TO THE VITAL BUS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 750 | Comment Type: IMPROVEMENT

LCO Number: 3.06.01.01 | Page Number: 3/4 6-1

Section Number: 4.6.1.1.a

Comment:

ADD FOOTNOTE MARKER "#" AFTER THE WORD "PENETRATIONS" IN THE FIRST LINE OF THE ACTION.

ADD NEW FOOTNOTE AT THE BOTTOM OF THE PAGE AS FOLLOWS:

#Valves CP-B3, CP-B7 and CM-B5 may be verified at least once per 31 days by manual remote keylock switch position.

Basis

THESE VALVES MAY NOT TAKE ADVANTAGE OF THE RELIEF OFFERED BY THE OTHER FOOTNOTE ON THIS PAGE DUE TO COMMITMENTS BY CP&L IN SUPPORT OF NUREG-0737 AS PER FSAR PAGES 6.2.5-6a & b. THIS FOOTNOTE CLARIFIES THAT A MONTHLY CHECK MUST BE DONE ON THE VALVES BUT THAT IT MAY BE OF THE SUPERVISORY CIRCUITS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 751 Comment Type: IMPROVEMENT

LCO Number: B 3/4.06.01.04 Page Number: B 3/4 6-1

Section Number: B 3/4.6.1.4

Comment:

REWORD THE BEGINING OF THE SECOND PARAGRAPH AS FOLLOWS:

.... line break event is 40.9 psig using a value of 1.9 psig for initial positive containment pressure. However, since the instrument.....

Basis

THIS CHANGE IS MADE TO MAKE THIS DISCUSSION MORE ACCURATE AND TO PROVIDE THE EXACT RESULTS OF THE LIMITING CALCULATION.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 752 Comment Type: ERROR

LCO Number: 3.04.04 Page Number: 3/4 4-11

Section Number: APPLICABILITY

Comment:

CHANGE THE APPLICABILITY TO "MODES 1, 2, 3 and 4*".

ADD A NEW FOOTNOTE AS FOLLOWS:
* Mode 4 when the temperature of all RCS cold legs is greater than 335 F."

Basis

THIS CHANGE IS NECESSARY TO INCLUDE THE 15 F OF MODE 4 WHERE THE PORV'S ARE REQUIRED TO BE OPERABLE, BUT ARE NOT AT THE REDUCED PRESSURE SETPOINTS DISCUSSED AS A PART OF THE COLD OVERPRESSURE SPECIFICATION.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 753

Comment Type: ERROR

LCO Number: 3.04.09.04

Page Number: 3/4 4-41

Section Number: FIGURE 3.4-4

Comment:

CHANGE THE Y-AXIS LABEL OF THE GRAPH TO "PORV SETPOINTS (psig)".

Basis

THIS CHANGE IS TO CORRECT A TYPOGRAPHICAL ERROR IN THE ORIGINAL INFORMATION SUPPLIED BY CP&L.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 754 | Comment Type: IMPROVEMENT

LCO Number: B 3/4 4-6 | Page Number: B 3/4 4-6 & 11

Section Number: B 3/4.4.9

Comment:

IN THREE PLACES, CHANGE "Figures 3.4-2 and 3.4-3" TO "Figures 3.4-3 and 3.4-2 and Table 4.4-6".

Basis

THESE CHANGES PROVIDE A MORE COMPLETE REFERENCE TO ALL OF THE PLACES WHICH PROVIDE HEATUP AND COOLDOWN LIMITATION DATA AND MAKE THE REFERENCES TO THE HEATUP AND COOLDOWN CURVES GRAMMATICALLY CORRECT.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 755

Comment Type: ERROR

LCO Number: 3.05.01

Page Number: 3/4 5-1

Section Number: 3.5.1.b

Comment:

CHANGE "97%" TO "96%".

Basis

THIS CHANGE IS NEEDED TO CORRECT AN ERROR IN ROUNDING OFF THE PERCENTAGE LIMIT. OUR CURRENT VALUE HAS THIS LIMIT ROUNDED IN A NON-CONSERVATIVE DIRECTION (BY 1.7 gal). THIS CHANGE ENSURES THAT THE ROUND OFF IS CONSERVATIVE.

CP&L Comments

NNPP Proof and Review Technical Specifications

Record Number: 756 Comment Type: ERROR

LCO Number: B 2.02.01 Page Number: B 2-5,6 & 7

Section Number: B 2.2.1

Comment:

ON PAGE B 2-5 & 6 CHANGE THE DESCRIPTION OF P-7 TO "...RATED THERMAL POWER or turbine impulse...."

ON PAGE B 2-7 IN THE PARAGRAPH ON TURBINE TRIP, INSERT "or a turbine impulse pressure at approximately 10% of full power equivalent" AFTER "RATED THERMAL POWER".

Basis

THIS CHANGE IS TO PROVIDE A MORE COMPLETE DESCRIPTION OF THE P-7 INTERLOCK CONSISTENT WITH OTHER BASES DISCUSSIONS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 757 | Comment Type: ERROR

LCO Number: 3.07.12 | Page Number: 3/4 7-41

Section Number: TABLE 3.7-6

Comment:

REVISE TABLE 3.7-6 PER THE ATTACHED MARKUP.

Basis

THE ELEVATION CHANGES ARE MADE TO PROVIDE ADDITIONAL INFORMATION OVERLOOKED ON PRIOR SUBMITTALS AND TO CORRECT TYPOGRAPHICAL ERRORS IN PREVIOUS CP&L SUBMITTALS.
ITEM 23 IS DELETED BECAUSE THE EXHAUST SILENCERS PERFORM NO SAFETY FUNCTION AND THEREFORE DO NOT NEED TO HAVE A TEMPERATURE LIMIT SPECIFICIED.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 758 Comment Type: IMPROVEMENT

LCO Number: 3.03.03.05 Page Number: 3/4 3-64 & 65

Section Number: TABLES

Comment:

ITEM 11 ON BOTH TABLE 3.3-9 AND 4.3-6 - ADD "(SR INDICATOR)" TO THE DESCRIPTION.

Basis

THE ADDITIONAL INFORMATION IS PROVIDED TO INDICATE EXACTLY WHICH INDICATOR IS REQUIRED. THERE ARE TWO INDICATORS TO COVER THE FULL RANGE OF THE WIDE RANGE FLUX MONITOR, BUT ONLY THE SOURCE RANGE IS REQUIRED FOR REMOTE SHUTDOWN ACTIVITIES.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 759 Comment Type: ERROR

LCO Number: 6.02.02 Page Number: 6-4

Section Number: FIGURE 6.2-2

Comment:

DELETE THE BLOCK FOR THE ADMINISTRATIVE SUPERVISOR WHICH REPORTS TO THE DIRECTOR PROGRAMS & PROCEDURES.

Basis

THIS POSITION NO LONGER EXISTS WITHIN THE SHNPP ORGANIZATION.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 760 Comment Type: IMPROVEMENT

LCO Number: 3.03.03.01 Page Number: 3/4 3-53

Section Number: TABLE 3.3-6

Comment:

CHANGE THE LAST SENTENCE OF ACTION 29 TO THE FOLLOWING:
.... System in the Pressurization Mode of Operation by opening one Emergency Outside Air Intake."

Basis

THIS CHANGE WILL PERMIT SOME AIR TO BE BROUGHT INTO THE CONTROL ROOM EVEN IF A TRAIN OF DETECTORS IS OUT OF SERVICE. PROTECTION WILL STILL BE PROVIDED BY THE OTHER TRAIN AND RECEIPT OF A CHLORINE SIGNAL WILL ALSO STILL CLOSE THE OPEN INLET VALVE.

CP&L Comments

HNPP Proof and Review Technical Specifications

Record Number: 761 | Comment Type: ERROR

LCO Number: 3.03.02 | Page Number: 3/4 3-34

Section Number: TABLE 3.3-4

Comment:

REVISE ITEM 9 OF TABLE 3.3-4 TO THE ATTACHED MARKUP.

Basis

THIS CHANGE IS MADE TO RETURN THE SPECIFICATION TO THE STANDARD FORMAT. THE DELETED DATA IS, AND SHOULD BE, PROCEDURALLY CONTROLED. THE DELETED DATA IS NOT RELATED TO ANY SAFETY ANALYSIS ASSUMPTION AND THEREFORE SHOULD NOT BE IN THE TECHNICAL SPECIFICATIONS.

THE CHANGE TO LESS THAN OR EQUAL TO 16 sec IN ITEM 9b IS TO COVER THE ANALYTICAL ASSUMPTIONS IN A MORE CONSERVATIVE MANNER BY SLIGHTLY REDUCEING THE TIME DELAY PRIOR TO THE DIESEL GENERATOR START.

CHANGES TO THE SETPOINTS ARE TO PROVIDE THE LATEST INFORMATION AVAILABLE TO US FROM THE A/E.

CP&L Comments

HNPP Proof and Review Technical Specifications

Record Number: 762 | Comment Type: IMPROVEMENT

LCO Number: 3.08.04.01 | Page Number: 3/4 8-19

Section Number: 4.8.4.1

Comment:

ADD A "*" AFTER THE WORD "OPERABLE".

ADD A NEW FOOTNOTE AT THE BOTTOM OF THE PAGE AS FOLLOWS:
* Fuses are provided for table completeness only; there are no surveillance requirements.

Basis

THIS FOOTNOTE IS CONSISTENT WITH THE CURRENT NRR PRACTICE AND IS INCLUDED JUST TO CLARIFY THE LACK OF SURVEILLANCE REQUIREMENTS FOR FUTURE AUDITORS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 763 Comment Type: IMPROVEMENT

LCO Number: 3.08.04.02 Page Number: 3/4 8-39

Section Number: 4.8.4.2

Comment:

ADD A "*" AFTER THE WORD "TEST".

ADD A NEW FOOTNOTE TO THE BOTTOM OF THE PAGE AS FOLLOWS:
* This test shall cover the bypass circuitry from the master bypass relay in the sequencer through operation of the local bypass relay.

Basis

THIS FOOTNOTE IS NEEDED FOR CLARIFICATION OF THE EXACT SCOPE OF THIS TEST. A QUESTION HAS BEEN RAISED CONCERNING WHETHER THE TEST SHOULD ALSO COVER THE WIRING DOWN TO THE ACTUAL VALVE; AND BACK TO THE SIGNAL TO START THE SEQUENCER. WE DO NOT FEEL THAT A TEST OF THIS DETAIL IS WARRANTED OR DESIRABLE ON THIS FREQUENCY. BASED ON A TELEPHONE DISCUSSION ON 7-18-86 WITH NRR PERSONNEL, WE BELIEVE THAT THIS FOOTNOTE ADEQUATELY EXPRESSES THE LEVEL OF TESTING ENVISIONED BY NRR IN THIS SPECIFICATION.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 764 | Comment Type: ERROR

LCO Number: 3.08.04.02 | Page Number: 3/4 8-40 & 42

Section Number: TABLE 3.8-2

Comment:

PAGE 3/4 8-40 VALVE NUMBER 1CS-472 - CHANGE FUNCTION TO "RCP SEAL WATER RETURN ISOLATION." PAGE 3/4 8-42 VALVES 1SW-97, 109, 98 & 110 - CHANGE THE FUNCTION TO "SW FROM FAN CLR .....".

Basis

ALL OF THESE CHANGES ARE TO CORRECT TYPOGRAPHICAL ERRORS IN THE ORIGINAL DATA SUPPLIED BY CP&L.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 765 Comment Type: IMPROVEMENT

LCO Number: 3.11.02.01 Page Number: 3/4 11-9

Section Number: TABLE 4.11-2

Comment:

EXTEND THE HORIZONTAL LINE IN THE CENTER OF THE ITEM THREE BLOCK OVER TO A POINT ABOVE THE LETTER "b".

Basis

THIS CHANGE IS NEEDED TO PROVIDE GREATER CLARITY TO THE TABLE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 766 | Comment Type: IMPROVEMENT

LCO Number: B 3/4.06.05 | Page Number: B 3/4 6-4

Section Number: B 2/4.6.5

Comment:

CHANGE THE TITLE OF THE SECTION TO "VACUUM RELIEF SYSTEM".

Basis

THIS CHANGE IS MADE TO PROVIDE CONSISTENCY WITH THE BODY OF THE SPECIFICATIONS.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 767 Comment Type: IMPROVEMENT

LCO Number: FIRE PROTECTION Page Number: VARIOUS

Section Number: FIRE PROTECTION

Comment:

DELETE THE FIRE PROTECTION SYSTEM SPECIFICATIONS
PER THE ATTACHED MARKUPS.

Basis

PER PREVIOUS CP&L LETTERS NLS-86-188 DATED JUNE 4,
1986 AND NLS-86-230 DATED JULY 22, 1986.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 768 Comment Type: IMPROVEMENT

LCO Number: 3.03.03.11 Page Number: 3/4 3-86

Section Number: ACTION b

Comment:

CHANGE THE SECOND SENTENCE OF THE ACTION STATEMENT TO THE FOLLOWING:

Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

Basis

THIS CHANGE IS MADE TO BE CONSISTENT WITH THE ACTION STATEMENT IN 3.3.3.10. THE TIME LIMITS WERE REMOVED FROM THE ACTION STATEMENTS BY THE NRC SOME TIME AGO. THIS IS NEEDED TO PROVIDE A LOGICAL REPORTING REQUIREMENT.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 769 | Comment Type: ERROR

LCO Number: 3.03.02 | Page Number: 3/4 3-37, 38,39

Section Number: TABLE 3.3-5

Comment:

CHNAGE THE FEEDWATER ISOLATION TIME FOR "7(3)" TO "12(3)" IN ITEMS 2a2, 3a2,4a2, AND 8b.

Basis

THIS CHANGE IS NECESSARY FOR CONSISTENCY WITH THE FSAR AND TECH SPEC TABLE 3.6-1 WHERE THE CLOSURE TIME FOR THE BYPASS LINES IS 10 sec.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 770　　　　Comment Type: IMPROVEMENT

LCO Number: 3.06.02.01　　　　Page Number: 3/4 6-11

Section Number: 4.6.2.1.b

Comment:

CHANGE THE SURVEILLANCE TO THE FOLLOWING:
By verifying, that in an indicated recirculation flow of at least 2150 gpm, each pump develops a discharge pressure of greater than or equal to 229 psig when tested pursuant to Specification 4.0.5.

Basis

THIS CHANGE IS NECESSARY TO SPECIFIY A FLOW SO THAT A POINT ON THE PUMP CURVE IS ADEQUATELY DETERMINED. THE PRESSURE HAS BEEN CHANGED TO CONFORM TO A POINT ON THE CURVE WHICH IS THE EASIEST TO TEST AND TO COMPLY WITH INSERVICE TEST CRITERIA.

CP&L Comments

HNPP Proof and Review Technical Specifications

Record Number: 771

Comment Type: ERROR

LCO Number: B 3/4.08.01

Page Number: B 3/4 8-1

Section Number: B 3/4.8.1

Comment:

IN THE SECOND LINE OF THE SECOND PARAGRAPH, CHANGE "five" TO "six".

Basis

ANOTHER TRANSMISSION LINE HAS RECENTLY BEEN PLACED INTO SERVICE.

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 772 Comment Type: ERROR

LCO Number: 6.02.01 Page Number: 6-3

Section Number: FIGURE 6.2-1

Comment:

DELETE THE BLOCK FOR "MANAGER ENGINEERING AND CONSTRUCTION SERVICES". ALSO, REMOVE THE "s" FOR THE TITLE OF THE MANAGER FUEL"S" DEPARTMENT.

Basis

THESE CHANGES ARE TO CORRECT A TYPOGRAPHICAL ERROR AND TO DELETE A POSITION WHICH NO LONGER EXISTS WITHIN THE ORGANIZATION

CP&L Comments

SHNPP Proof and Review Technical Specifications

Record Number: 773 Comment Type: IMPROVEMENT

LCO Number: 3.03.02 Page Number: 3/4 3-41 etc.

Section Number: TABLE 4.3-2

Comment:

ITEMS 1b, AND 4b - UNDER THE "MODES FOR WHICH SURVEILLANCE IS REQUIRED" COLUMN, ADD MODE 4. ITEM 4e - UNDER THE "MODE FOR WHICH SURVEILLANCE IS REQUIRED" COLUMN, CHANGE TO 3**, 4**. UNDER THE TABLE NOTATIONS, ADD NEW NOTE - ** Trip function automatically blocked above P-11 and may be blocked below P-11 when safety injection or low steam lime pressure is not blocked.

Basis

THESE CHANGES ARE NEEDED FOR CONSISTENCY WITH MODE CHANGES TO ITEMS 1c AND 4e OF TABLE 3.3-3 WHICH WERE REQUESTED BY NRR REVIEWS IN APRIL 1986 AND AGREED TO BY CP&L. THE NOTE SIMPLY AGREES WITH THE PREVIOUS TABLE ON THESE MODES.

CP&L Comments

NPP Proof and Review Technical Specifications

Record Number: 774

Comment Type: ERROR

LCO Number: 6.05.04.03

Page Number: 6-15

Section Number: 6.5.4.3

Comment:

CHANGE THE TITLE TO SENIOR EXECUTIVE VICE PRESIDENT-POWER SUPPLY AND ENGINEERING AND CONSTRUCTION.

Basis

TYPOGRAPHICAL

CP&L Comments

NPP Proof and Review Technical Specifications

Record Number: 775 | Comment Type: ERROR

LCO Number: 3.07.12 | Page Number: 3/4 7-41

Section Number: TABLE 3.7-6

Comment:

ITEM 25 - CHANGE THE MAX TEMP TO "118".
ITEM 26 - CHANGE THE MAX TEMP TO "116".

Basis

THESE TEMPERATURES HAVE BEEN REVISED AS A RESULT OF A REEVALUATION OF THE ENVIRONMENTAL QUALIFICATION OF THE EQUIPMENT IN THE AREA.

CP&L Comments

HNPP Proof and Review Technical Specifications

Record Number: 776 Comment Type: ERROR

LCO Number: 3.06.01.05 Page Number: 3/4 6-7

Section Number: 4.6.1.5

Comment:

CHANGE THE LOCATIONS TO THE FOLLOWING:

a. Elevation 290 ft
b. Elevation 240 ft
c. Elevation 230 ft

Basis

THESE CHANGES ARE NECESSARY DUE TO A RECENT MODIFICATION TO THE TEMPERATURE MONITORS WHICH CHANGED THE LOCATIONS.

CP&L Comments

NPP Proof and Review Technical Specifications

Record Number: 777

Comment Type: IMPROVEMENT

LCO Number: 3.09.06

Page Number: 3/4 9-7

Section Number: 4.9.6.1

Comment:

CHANGE "when the refueling machine load exceeds" TO "at less than or equal to".

Basis

THIS CHANGE IS NECESSARY TO ENSURE THAT THE LOAD CUTOFF IS SET AT OR BELOW 2700 lbs., NOT WHEN THE LOAD EXCEEDS 2700 lbs.

CP&L Comments

HNPP Proof and Review Technical Specifications

Record Number: 778　　　　Comment Type: ERROR

LCO Number: NRC TYPO's　　　　Page Number: SEE LIST

Section Number:

Comment:

CHANGES HAVE BEEN MADE TO THE FOLLOWING PAGES TO CORRECT TYPOGRAPHICAL ERRORS MADE IN THE TYPING OF THE FINAL DRAFT TECH SPECS.

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Basis

TYPOGRAPHICAL ERRORS

Attachment 2 to NLS-86-273

Markup of TS pages
reflecting Final Draft Comments

FINAL DRAFT TECHNICAL SPECIFICATIONS

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(4022GAS/crs)

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FINAL DRAFT

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TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.099;

K_2 = 0.0182/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 20 s, τ_5 = 4 s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	1.086,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

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Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER ~~with~~ OR turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER ~~with~~ OR a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 91.7% of nominal full loop flow. Above P-8

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 91.7% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.627×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 38.5%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER ~~with~~ OR a turbine impulse chamber pressure at approximately

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-7.

OR A TURBINE IMPULSE PRESSURE AT APPROXIMATELY 10% OF FULL POWER EQUIVALENT.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pump, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	15*
d. Pressurizer Pressure--Low	3	2	2	1, 2, 3#	15*
e. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15*

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Main Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, ~~4~~	21
c. Containment Pressure--High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low initiating functions and requirements.				
e. Negative Steam Line Pressure Rate--High	3/steam line	2 in any steam line	2/steam line	3***, 4***	15*
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2	19*
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	~~22~~ 23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Steam Generator Water Level--Low-Low					
1) Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. in any stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
2) Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. in any 2 stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for Loss of Offsite Power initiating functions and requirements.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	1/pump	1/pump	1/pump	1, 2	18

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, ~~restore the inoperable channel to OPERABLE status within 48 hours o~~r declare the ~~associated valve~~ AFFECTED COMPONENT inoperable and take the ~~ACTION required by Specification 3.7.1.5~~. APPROPRIATE ACTION REQUIRED PER THE SPECIFIC COMPONENT'S SPECIFICATION.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - During CORE ALTERATIONS or movement of irradiated fuel within containment, comply with the ACTION statement of Specification 3.9.9.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE ~~DROPOUT~~	~~PICKUP~~
9. Loss-of-Offsite Power						
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	≥ 4830 volts with a ≤ 1.0 second time delay.	~~≥ 4900 volts and~~ ≥ 4692 volts with a time delay ~~between 0.95~~ and ≤1.5 seconds.	~~N.A.~~
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	≥ ~~6424~~ 6420 volts with a ≥ ~~15.6~~ 16 second time delay (with Safety Injection).	≥ ~~6400~~ 6392 volts with a time delay ~~between 14 and~~ ≤ 18 seconds (with Safety Injection).	~~≤ 6478 volts~~
				≥ ~~6424~~ 6420 volts with a ≤ 54.0 second time delay (without Safety Injection).	≥ ~~6400~~ 6392 volts with a ≤ 60 second time delay (without Safety Injection).	~~≤ 6478 volts~~
10. Engineered Safety Features Actuation System Interlocks						
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≥ 2000 psig	≥ 1986 psig	
Not P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2014 psig	
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 550.6°F	

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Containment Ventilation Isolation	N.A.
e. Steam Line Isolation	N.A.
f. Reactor Trip	N.A.
g. Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq$ X$^{(3)}$/2
3) Containment Phase "A" Isolation	$< 62^{(2)}/72^{(1)}$
4) Containment Ventilation Isolation	$\leq 4.75^{(6)}$
5) Auxiliary Feedwater Motor-Driven Pumps	≤ 60
6) Emergency Service Water Pumps	$\leq 32^{(1)}/22^{(8)}$
7) Containment Fan Coolers	$\leq 27^{(1)}/17^{(8)}$
8) Control Room Isolation	N.A.
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq$ X$^{(3)}$/2
3) Containment Phase "A" Isolation	$< 62^{(2)}/72^{(1)}$
4) Containment Ventilation Isolation	$\leq 4.75^{(6)}$

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low (Continued)	
a. Safety Injection (ECCS) (Continued)	
5) Auxiliary Feedwater Motor-Driven Pumps	≤ 60
6) Emergency Service Water Pumps	≤ 32(1)/22(8)
7) Containment Fan Coolers	≤ 27(1)/17(8)
8) Control Room Isolation	N.A.
4. Main Steam Line Pressure--Low	
a. Safety Injection (ECCS)	≤ 12(5)/22(4)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 12(3)
3) Containment Phase "A" Isolation	< 62(2)/72(1)
4) Containment Ventilation Isolation	≤ 4.75(6)
5) Auxiliary Feedwater Motor-Driven Pumps	≤ 60
6) Emergency Service Water Pumps	≤ 32(1)/22(8)
7) Containment Fan Coolers	≤ 27(1)/17(8)
8) Control Room Isolation	N.A.
b. Steam Line Isolation	≤ 12
5. Containment Pressure--High-3	
a. Containment Spray	≤ 18.5(8)/32.2(1)
b. Phase "B" Isolation	≤ 22.5(1)/12(2)
6. Containment Pressure--High-2	
Steam Line Isolation	≤ 12
7. Negative Steam Line Pressure Rate -- High	
Steam Line Isolation	≤ 12

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YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.10.4 The yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses given in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.4 Each of the yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to assure all required equipment is at the hose house,

b. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater,

   2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

   3. Performing a flow check of each hydrant to verify its OPERABILITY.

   4. Visually inspecting each yard hydrant and verifying that the hydrant is dry and is not damaged (to be performed during September, October, or November)

   5. Visually inspecting each hydrant and verifying that it is not damaged (to be performed during March, April or May)

~~TABLE 3.7-4 (Continued)~~

~~FIRE HOSE STATIONS~~

LOCATION[1]	ELEVATION	HOSE RACK NO.
RAB	261	261-Jz-43
RAB	261	261-Fw-43
RAB	305	305-C-39
RAB	305	305-I-41
RAB	305	305-Fw-43
RAB	236	236-JZ-45
RAB	286	286-JV-45
RAB	286	286-FW-44
RAB	305	305-I-45
RAB	324	324-I-41
RAB	324	324-I-45
FHB	236	236-L-41
FHB	236	236-L-45
FHB	261	261-L-41
FHB	261	261-L-45
FHB	286	286-L-27
FHB	286	286-N-30
FHB	286	286-L-43
FHB	286	286-N-51
FHB	286	286-L-65
FHB	286	286-N-71
FHB	286	286-L-75y
FHB	216	216-L-41
FHB	216	216-L-45
FHB	216	216-L-71
FHB	236	236-L-71
FHB	261	261-N-73
FHB	261	261-M-75y
DGB	261	261-C-2
DGB	261	261-C-4
DGB	261	261-B-1
DGB	261	261-B-2
DFOSB	261	1-4H NNS**
DFOSB	261	1-4V NNS**
ESWIS	261	1-4AJ NNS**
ESWISS	261	1-4AI NNS**

**Yard Hydrant

[1] CNMT - Containment Building
RAB - Reactor Auxiliary Building
ESWIS - Emergency Service Water Intake Structure
DFOSB - Diesel Fuel Oil Storage Building
FHB - Fuel Handling Building
DGB - Diesel Generator Building
ESWISS - Emergency Service Water Intake Screening Structure

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TABLE 3.7-5

~~YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES~~ DELETED

LOCATION		HYDRANT NUMBER
Emergency Service Water Intake Structure		1-4AJ-NNS
Emergency Service Water Screening Structure		1-4AI-NNS
Diesel Generator Building	North Side	1-4B-NNS
	South Side	1-4A-NNS
Diesel Fuel Oil Storage Tank Building	East Side	1-4V-NNS
	West Side	1-4H-NNS

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3/4.7.11 ~~FIRE RATED ASSEMBLIES~~ DELETED

LIMITING CONDITION FOR OPERATION

3.7.11 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals shall be OPERABLE.

APPLICABILITY: Whenever the equipment in the area is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door is closed at least once per 7 days,

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours, and a functional test is performed at least once per 18 months, and

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ ~~7~~[3] 12
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. Loss-of-Offsite Power	
Motor-and Turbine-Driven Auxiliary Feedwater Pumps	≤ 60
11. Trip of All Main Feedwater Pumps	
Motor-Driven Auxiliary Feedwater Pumps	N.A.
12. Steam Line Differential Pressure--High Coincident with Main Steam Line Isolation Signal	
Isolate Auxiliary Feedwater to the Affected Steam Generator	≤ 41
13. RWST Level--Low-Low	
a. Safety Injection Switchover to Containment Sump Coincident with Safety Injection	≤ 32
b. Safety Injection Switchover to Containment Sump Coincident With Containment Spray	≤ 103
14. Containment Radioactivity-High	
a. Normal Containment Purge Isolation	≤ 3.5[7]
b. Preentry Containment Purge Isolation	≤ 15[7]

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure--High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
(2) Preentry Purge Detector	See Table 4.3-3, Item 1b2, for surveillance requirements.							
c) Airborne Particulate Radioactivity								
(1) RCS Leak Detection (normal purge)	See Table 4.3-3, Item 1C1, for surveillance requirements.							
(2) Preentry Purge Detector	See Table 4.3-3, Item 1C2, for surveillance requirements.							
5) Manual Phase A Isolation	See Item 3.a.1) above for Manual Phase A Isolation Surveillance Requirements.							
4. Main Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Main Steam Line Isolation (Continued)								
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Surveillance Requirements.							
e. Negative Steam Line Pressure Rate--High	S	R	M	N.A.	N.A.	N.A.	N.A.	3**, 4**
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level--High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
d. Safety Injection Start Motor-Driven Pump	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Power Surveillance Requirements.							

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) The Surveillance Requirements of Specification 4.9.9 apply during CORE ALTERATIONS or movement of irradiated fuel in containment.

(3) Except for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 which shall be tested at least once per 18 months and during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days.

* Setpoint verification not required.

\# During CORE ALTERATIONS or movement of irradiated fuel in containment.

** TRIP FUNCTION AUTOMATICALLY BLOCKED ABOVE P-11 AND MAY BE BLOCKED BELOW P-11 WHEN SAFETY INJECTION OR LOW STEAM LINE PRESSURE IS NOT BLOCKED.

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

** With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

*** Each channel consists of 3 detectors with 1 of 3 logic. A channel is OPERABLE when 1 or more of the detectors are OPERABLE.

\# Setpoint to be less than or equal to three times detector background at RATED THERMAL POWER.

\## Required OPERABLE whenever pre-entry purge system is to be used.

ACTION STATEMENTS

ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, suspend all operations involving movement of fuel within the storage pool or crane operations over the storage pool.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour initiate isolation of the respective air intake. With no outside air intakes available, maintain operation of the Control Room Emergency Filtration System in the ~~Recirculation~~ PRESSURIZATION Mode of Operation BY OPENING ONE EMERGENCY OUTSIDE AIR INTAKE.

ACTION 30 - With less than the minimum channels OPERABLE requirement, pre-entry purge operations shall be suspended and the containment pre-entry purge makeup and exhaust valves shall be maintained closed.

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TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant System Hot-Leg Temperature	ACP*	2	~~1~~ 2
2. Reactor Coolant System Cold-Leg Temperature	ACP*	2	~~1~~ 2
3. Pressurizer Pressure	ACP*	2	1-SSA Channel**
4. Pressurizer Level	ACP*	2	1-SSA Channel**
5. Steam Generator Pressure (Note 1)	ACP*	1/Steam Generator	1/Steam Generator
6. Steam Generator Water Level--Wide Range (Note 1)	ACP*	1/Steam Generator	1/Steam Generator
7. Residual Heat Removal Flow ~~(Note 2)~~	ACP*	2	1 (Note 2)
8. Auxiliary Feedwater Flow (Note 1)	ACP*	1/Steam Generator	N.A. (Note 3)
9. Condensate Storage Tank Level	ACP*	2	1-SSA Channel**
10. Reactor Coolant System Pressure-Wide Range	ACP*	2	1-SSA Channel**
11. Wide-Range Flux Monitor (SR Indicator)	ACP*	1	1-SSA Channel**
12. Charging Header Flow	ACP*	1	1-SSA Channel**
13. a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP or	ACP*	1	1-SSA Channel**
b. Auxiliary Feedwater Turbine Speed	ACP*	1	1-SSA Channel**
14. Boric Acid Tank Level	ACP*	1	1-SSA Channel**

*ACP = Auxiliary Control Panel
**SSA = Safe Shutdown Analysis

Note 1 - Steam Generators A&B Only
Note 2 - RHR Train B Only
Note 3 - Steam Generator Water Level is used

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant System Hot-Leg Temperature	M	R
2. Reactor Coolant System Cold-Leg Temperature	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Pressure	M	R
6. Steam Generator Water Level--Wide Range	M	R
7. Residual Heat Removal Flow	M	R
8. Auxiliary Feedwater Flow	M	R
9. Condensate Storage Tank Level	M	R
10. Reactor Coolant System Pressure--Wide Range	M	R
11. Wide-Range Flux Monitor (SR INDICATOR)	M	Q
12. Charging Header Flow	M	R
13. a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP	M	R
b. Auxiliary Feedwater Turbine Speed	M	R
14. Boric Acid Tank Level	M	R

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LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Required Number of Channels shown in Table 3.3-10, except for the pressurizer safety valve position indicator or the sub-cooling margin monitor, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the pressurizer safety valve position indicator, or the sub-cooling margin monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

c. With the number of OPERABLE channels for the radiation monitors, the pressurizer safety valve position indicator*, or the sub-cooling margin monitor#, less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

---

* The alternate method shall be a check of SAFETY VALVE PIPING ~~[illegible]~~ temperatures and evaluation to determine position. ←

The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

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~~FIRE DETECTION INSTRUMENTATION~~ DELETED

~~LIMITING CONDITION FOR OPERATION~~

3.3.3.8 ~~As a minimum, the fire detection instrumentation for each fire~~ detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5).

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST*. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST* during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

* Setpoints for heat and flame detectors need not be verified. Setpoints for smoke detectors shall be verified on every other test interval.

TABLE 3.3-11

~~FIRE DETECTION INSTRUMENTATION~~ DELETED

ZONE	INSTRUMENT LOCATION	ELEVATION (FT)	~~TOTAL NUMBER OF INSTRUMENTS~~ HEAT (A/B)*	FLAME (A/B)*	SMOKE (A/B)*
1.0 Containment Building**					
1-C-1-RCP-1A	Reactor Coolant Pump 1A	256.33	12/0	---	---
1-C-1-RCP-1B	Reactor Coolant Pump 1B	256.33	12/0	---	---
1-C-1-RCP-1C	Reactor Coolant Pump 1C	256.33	12/0	---	---
1-C-1-CHFA	Airborne Radioactivity Removal Unit 1A	221.0	0/5	---	---
1-C-1-CHFB	Airborne Radioactivity Removal Unit 1B	221.0	0/9	---	---
1-C-3-EPA	Electrical Penetration Area 1A	261.0	0/12	---	12/0
1-C-3-EPB	Electrical Penetration Area 1B	261.0	0/12	---	12/0
1-C-BAL	Elevator Machine Room	302.0	---	---	2/0
1-C-1	Containment Fan Coolers	236.0	0/4	---	---
1-C-1	Containment 263 & Press. Cubicle No. 309	263/309	0/24	---	---
2.0 Reactor Auxiliary Building					
1-A-1-PA	RHR Pump Room 1A	190.0	0/11	---	---
1-A-1PB	RHR Pump Room 1B	190.0	0/11	---	---
1-A-2MP	Misc. Pumps & Equipment	216.0	0/28	---	32/0
1-A-3-PB	Auxiliary Feedwater Pumps, Component Cooling Water Pumps & Heat Exchangers	236.0	0/56	---	59/0

*(A/B) A = The number of early warning fire detectors.
B = The number of detectors used for actuation of fire suppression systems.

**The fire detection instruments located within the Containment Building are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

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~~TABLE 3.3-11 (Continued)~~

~~FIRE DETECTION INSTRUMENTATION~~

ZONE	INSTRUMENT LOCATION	ELEVATION (FT)	~~TOTAL NUMBER OF INSTRUMENTS~~ HEAT (A/B)*	FLAME (A/B)*	SMOKE (A/B)*
2.0	Reactor Auxiliary Building (Continued)				
1-A-3-COME	Decontamination Area & Corridor Cable Trays	236.0	0/10	---	14/0
1-A-3-COME	Letdown Heat Exchanger & Corridor Cable Trays	236.0	0/6	---	18/0
1-A-3-COR	Corridor Cable Trays	236.0	0/4	---	---
1-A-3-COMI	Recycle Holdup Tank Area & Corridor Cable Trays	236.0	0/10	---	22/0
1-A-4-CHLR	HVAC Chiller Equipment Area & Cable Trays	261.0	0/44	---	44/0
1-A-4-COM-B	Boric Acid Equipment Area & Corridor Cable Trays	261.0	0/12	---	13/0
1-A-4-COM-E	Corridor Cable Trays	261.0	0/7	---	12/0
1-A-4-COM-I	Corridor Cable Trays	261.0	0/3	---	7/0
1-A-4-CHFA	Charcoal Filter Room 1A	261.0	0/5	---	10/0
1-A-4-CHFB	Charcoal Filter Room 1B	261.0	0/4	---	8/0
1-A-EPA	Electrical Penetration Area SA	261.0	0/15	---	15/0
1-A-EPB	Electrical Penetration Area SB	261.0	0/15	---	15/0
1-A-5-HVA	HVAC Room 1A	286.0	---	---	14/0
1-A-5-HVB	HVAC Room 1B	286.0	---	---	15/0
1-A-SWGRA	Switchgear Room A	286.0	---	---	18/0
1-A-SWGRB	Switchgear Room B	286.0	---	---	17/0
1-A-BATA	Battery Room 1A	286.0	---	---	2/0
1-A-BATB	Battery Room 1B	286.0	---	---	2/0
1-A-CSRA	Cable Spreading Room A	286.0	0/27	---	27/0

~~Table 3.3-11 (Continued)~~

ZONE	INSTRUMENT LOCATION	ELEVATION (FT)	~~TOTAL NUMBER OF INSTRUMENTS~~ HEAT (A/B)*	FLAME (A/B)*	SMOKE (A/B)*
2.0	Reactor Auxiliary Building (Continued)				
1-A-CSRB	Cable Spreading Room B	286.0	0/15	---	15/0
1-A-ACP	Auxiliary Control Panel	286.0	---	---	2/0
1-A-CSRA	RIC Room A	286.0	---	---	2/0
12-A-6-RT1	Terminal Cabinet Room	305.0	---	---	14/0
12-A-6-RCC1	Rod Control Cabinets Room	305.0	---	---	9/0
12-A-6-CR1	Main Control Room Area	305.0	---	---	20/0
12-A-6-APR1	Auxiliary Relay Panels	305.0	---	---	6/0
12-A-6-CR1	Control Room Panels	305.0	---	---	9/0
12-A-6-PICR1	Process Instruments & Control Racks	305.0	---	---	8/0
12-A-6-HV7	HVAC Equipment Room	305.0	0/8	---	24/0
3.0	Fuel Handling Building				
5-F-2-FPC	Fuel Pool Cooling Pumps & Heat Exchangers	236.0	0/16	---	2/0
5-F-3-CHFA	Emergency Exhaust Charcoal Filter A	261.0	0/8	---	8/0
5-F-3-CHFB	Emergency Exhaust Charcoal Filter B	261.0	0/8	---	8/0
5-F-3-CHF-BAL	Emergency Exhaust Balance	261.0	---	---	4/0
5-F-3-DMNZ	General Area	261.0	---	---	12/0
4.0	Diesel Generator Building				
1-D-1-DGA-RM	Diesel Generator 1A	261.0	0/10	4/0	---
1-D-1-DGB-RM	Diesel Generator 1B	261.0	0/10	4/0	---
1-D-1-DGA-ASU	Diesel Generator Air Starting Unit 1A	261.0	2/0	---	---

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~~TABLE 9.5-11 (Continued)~~

ZONE	INSTRUMENT LOCATION	ELEVATION (FT)	~~TOTAL NUMBER OF INSTRUMENTS~~ HEAT (A/B)*	FLAME (A/B)*	SMOKE (A/B)*
4.0	Diesel generator Building (Continued)				
1-D-1-DGB-ASU	Diesel Generator Air Starting Unit 1B	261.0	2/0	---	---
1-D-1-DGA-TK	Diesel Fuel Oil Day Tank 1A	280.0	0/2	---	---
1-D-1-DGB-TK	Diesel Fuel Oil Day Tank 1B	280.0	0/2	---	---
1-D-1-DGA-ER	Diesel Generator MCC & Control Panel 1A	261.0	---	---	2/0
1-D-1-DGB-ER	Diesel Generator MCC & Control Panel 1B	261.0	---	---	2/0
1-D-3-DGA-ES	Diesel Exhaust Silencer 1A	292.0	---	2/0	---
1-D-3-DGB-ES	Diesel Exhaust Silencer 1B	292.0	---	2/0	---
5.0	Diesel Oil Storage Tank Area				
1-0-PA	Diesel Fuel Oil Pump Room 1A	242.25	0/2	2/0	---
1-0-PB	Diesel Fuel Oil Pump Room 1B	242.25	0/2	2/0	---
5-0-BAL	Diesel Fuel Oil Storage Tank Area--Balance	242.25	---	7/0	---
6.0	Emergency Service Water Intake Structure				
12-1-ESWPA	Electrical Equipment Room SA	251.7/ 262.0	---	---	8/0
	Pump Room SA	262.0	---	2/0	---
12-1-ESWPB	Electrical Equipment Room SB	251.7/ 262.0	---	---	8/0
	Pump Room SB	262.0	---	2/0	---

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately (1) suspend the release of radioactive gaseous effluents monitored by the affected channel or (2) declare the channel inoperable and take ACTION as directed by b. below.

b. With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. ~~Restore the inoperable~~ EXERT BEST EFFORTS TO RETURN THE instrument~~ation~~ to OPERABLE status within ~~the time specified in the~~ 30 DAYS AND IF UNSUCCESSFUL ~~ACTION, or~~ explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected ~~within the time specified~~ IN A TIMELY MANNER.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and a DIGITAL CHANNEL OPERATIONAL TEST or an ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. WASTE PROCESSING BUILDING VENT STACK 5					
a.1 Noble Gas Activity Monitor (PIG)	D	M	R(3)	Q(1)	*
a.2 Noble Gas Activity Monitor (WRGM)	D	M	R(3)	Q~~(1)~~(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
5. WASTE PROCESSING BUILDING VENT STACK 5A					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, ~~and~~ 3, AND 4*

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable as a result of causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve.

c. With two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With all three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

e. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b., c. or d. above, as appropriate, for the isolated PORV(s).

f. The provisions of Specification 3.0.4 are not applicable.

* MODE 4 WHEN THE TEMPERATURE OF ALL RCS COLD LEGS IS GREATER THAN 335°F.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment Airborne Gaseous or Particulate Radioactivity Monitor at least once per 12 hours;

b. Monitoring the containment sump inventory and ~~discharge~~ FLOW MONITORING SYSTEM at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 335°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

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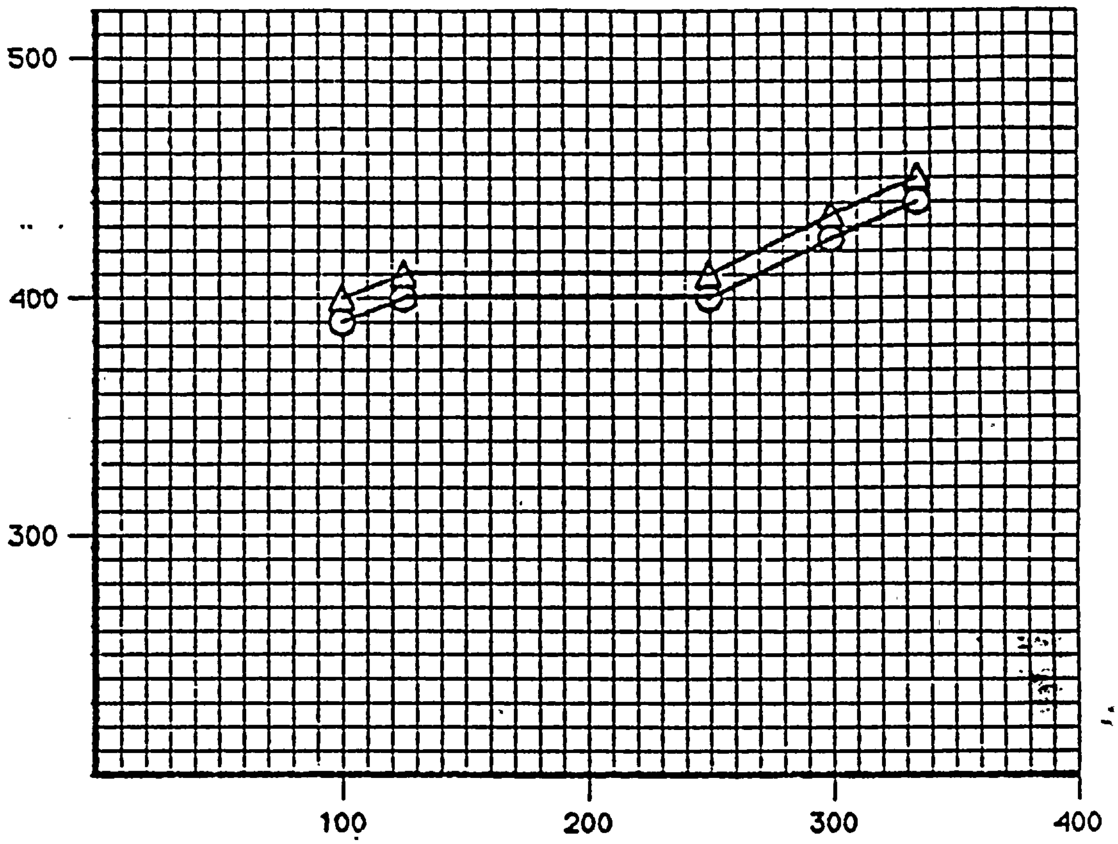

RCS TEMP OF	LOW PORV * PSIG O	HIGH PORV * PSIG Δ
< 100	390	400
125	400	410
250	400	410
300	425	435
335	440	450

* VALUES BASED ON 4 EFPY REACTOR VESSEL DATA AND CONTAINS MARGINS OF −10°F AND +60 PSIG FOR POSSIBLE INSTRUMENT ERROR

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open WITH POWER SUPPLY CIRCUIT BREAKER OPEN.

b. A contained borated water volume of between 66 and ~~92%~~ 96 indicated level.

c. A boron concentration of between 2000 and 2200 ppm, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*~~Pressurizer~~ RCS pressure above 1000 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

~~HPSI SYSTEM EBASCO Valve No (FSAR)~~	HPSI SYSTEM EBASCO Valve No.	HPSI SYSTEM CP&L Valve No.
~~2SI-V16SA-1~~	2SI-V440SA-1	1SI-5
~~2SI-V22SB-1~~	2SI-V439SB-1	1SI-6
~~2SI-V28SA-1~~	2SI-V438SA-1	1SI-7
~~2SI-V62SA-1~~	2SI-V437SA-1	1SI-69
~~2SI-V60SB-1~~	2SI-V436SB-1	1SI-70
~~2SI-V74SA-1~~	2SI-V435SA-1	1SI-71
~~2SI-V30SA-1~~	2SI-V434SA-1	1SI-101
~~2SI-V44SB-1~~	2SI-V433SB-1	1SI-102
~~2SI-V50SA-1~~	2SI-V432SA-1	1SI-103
~~2SI-V83SA-1~~	2SI-V431SA-1	1SI-124
~~2SI-V89SB-1~~	2SI-V430SB-1	1SI-125
~~2SI-V95SA-1~~	2SI-V429SA-1	1SI-126

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For charging/safety injection pump lines, with a single pump running:

    a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 379 gpm, and

    b) The total pump flow rate is less than or equal to 650 gpm.

2. For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

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3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations*# not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 41 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

VALVES CP-B3, CP-B7 AND CM-B5 MAY BE VERIFIED AT LEAST ONCE PER 31 DAYS BY MANUAL REMOTE KEYLOCK SWITCH POSITION

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

SURVEILLANCE REQUIREMENTS (Continued)

period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the following equation:

   $|L_c - (L_{am} + L_o)| \leq 0.25\ L_a$, where L_{am} is the measured Type A test leakage and L_o is the superimposed leak;

2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3. Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, at intervals no greater than 24 months except for tests in volving:

1. Air locks,

2. Containment purge makeup and exhaust isolation valves with resilient material seals,

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge makeup and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.~~3~~ 2 ~~or 4.6.1.7.4, as applicable~~;

g. The provisions of Specification 4.0.2 are not applicable.

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LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* EXCEPT DURING ENTRY TO REPAIR AN INOPERABLE INNER DOOR, FOR A CUMULATIVE TIME NOT TO EXCEED 1 HOUR PER YEAR.

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CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours, to be within the limit:

Location

a. Elevation 290 ft ~~- 3 locations~~
b. Elevation 240 ft
c. Elevation 230 ft

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CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1. ←

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1. Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined, during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2), by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports. Any abnormal degradation of the containment vessel structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

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3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on AN INDICATED recirculation flow OF AT LEAST 2150 gpm, each pump develops a discharge pressure of greater than or equal to ~~245~~ 229 psig when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal and

2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

3. Verifying that COINCIDENT WITH AN INDICATION OF CONTAINMENT SPRAY PUMP RUNNING, each automatic ~~recirculation~~ valve from the sump AND RWST actuates TO ITS APPROPRIATE POSITION FOLLOWING on an RWST Lo-Lo test signal ~~and safety injection test signal.~~

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

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SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 2736 and 2912 gallons of between 28% and 30% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and
2. Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a containment spray or containment isolation phase A test signal as applicable; and

d. At least once per 5 years by verifying each eductor flow rate is BETWEEN 19.5 AND 20.5 ~~greater than or equal to 17~~ gpm, using the RWST as the test source containing at least ~~100,000~~ 436,000 gallons of water.

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
78C	1SP-59 (SP-V1)	PRESSURIZER STEAM SAMPLE	<60	2
78C	1SP-60 (SP-V2)	PRESSURIZER STEAM SAMPLE	<60	2
78D	1SP-78 (SP-V113)	ACCUMULATOR SAMPLE	<60	2
78D	1SP-81 (SP-V114)	ACCUMULATOR SAMPLE	<60	2
78D	1SP-84 (SP-V115)	ACCUMULATOR SAMPLE	<60	2
78D	1SP-85 (SP-V116)	ACCUMULATOR SAMPLE	<60	2
80	1IA-~~210~~ 819 (IA-V192)	INSTRUMENT AIR SUPPLY	<60	N/A
88	1SP-201 (SP-V406)	LIQUID SAMPLE RETURN FROM PASS SKID #1	5	3
88	1SP-200 (SP-V407)	LIQUID SAMPLE RETURN FROM PASS SKID #1	5	3
91	1SW-240 (SW-B89)	SERVICE WATER FROM NNS FAN COILS	<60	2
91	1SW-242 (SW-B90)	SERVICE WATER FROM NNS FAN COILS	<60	2
92	1SW-231 (SW-B88)	SERVICE WATER TO NNS FAN COILS	<60	2
105	1FP-347 (FP-B1)	FIRE WATER SPRINKLER SUPPLY	<60	2
108	1AF-155 (AF-V162)	AUX. F.W. TO S/G A (HYDRAZINE)	10	1,2,6
108	1AF-153 (AF-V163)	AUX. F.W. TO S/G A (AMMONIA)	10	1,2,6

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
39	1CC-249 (CC-V191)	CCW FROM RCP THERMAL BARRIERS	10	N/A
39	1CC-251 (CC-V190)	CCW FROM RCP THERMAL BARRIERS	10	N/A
3. SAFETY INJECTION ACTUATION				
8	1CS-238 (CS-V610)	CVCS NORMAL CHARGING	10	N/A
51	1BD-11 (BD-V11)	S/G A BLOWDOWN	<60	1,2,6
52	1BD-30 (BD-V15)	S/G B BLOWDOWN	<60	1,2,6
53	1BD-49 (BD-V19)	S/G C BLOWDOWN	<60	1,2,6
54	1SP-217 (SP-V120)	S/G A SAMPLE	<60	1,2,6
55	1SP-222 (SP-V121)	S/G B SAMPLE	<60	1,2,6
56	1SP-227 (SP-V122)	S/G C SAMPLE	<60	1,2,6
4. CONTAINMENT VENTILATION ISOLATION				
57	CP-B1 (CP-B1)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5
57	CP-B3 (CP-B3)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	2,5
57	CP-B4 (CP-B4)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	2,5
57	CP-B2 (CP-B2)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5
58	CP-B7 (CP-B7)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	2,5

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
58	CP-B5 (CP-B5)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5
58	CP-B8 (CP-B8)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	2,5
58	CP-B6 (CP-B6)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5
59	CB-B1 (CB-B1)	CONTAINMENT VACUUM RELIEF	5	3
98	CB-B2 (CB-B2)	CONTAINMENT VACUUM RELIEF	5	3
5. CONTAINMENT SPRAY ACTUATION				
23	1CT-50 (CT-V21)	CONTAINMENT SPRAY	N/A	3
24	1CT-88 (CT-V43)	CONTAINMENT SPRAY	N/A	3
6. MAIN ST~~R~~EAM LINE ISOLATION				
3	1MS-80 (MS-V1)	MSIV (S/G A)	5	1,4
3	1MS-81 (MS-F1)	MSIV BYPASS	10	1,2,3,6
3	1MS-231 (MS-V59)	MS DRAIN TO CONDENSER	<60	1,2,6
2	1MS-82 (MS-V2)	MSIV (S/G B)	5	1,4
2	1MS-83 (MS-F2)	MSIV BYPASS	10	1,2,3,6
2	1MS-266 (MS-V60)	MS DRAIN TO CONDENSER	<60	1,2,6
1	1MS-84 (MS-V3)	MSIV (S/G C)	5	1,~~3~~4

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
32	1SW-110 (SW-B50)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1,6
17	1SI-3 (SI-V505)	SI TO HIGH HEAD COLD LEG	N/A	3
17	1SI-4 (SI-V506)	SI TO HIGH HEAD COLD LEG	N/A	3
2	1MS-70 (MS-V8)	MAIN STEAM B TO AUXILIARY F.W. TURBINE	N/A	1,3,6
1	1MS-72 (MS-V9)	MAIN STEAM C TO AUXILIARY F.W. TURBINE	N/A	1,3,6
63	CM-B5 (CM-B5)	H_2 PURGE EXHAUST	N/A	3
10. MANUAL VALVES				
17	1SI-43 (SI-V30)	SI-HIGH HEAD TO COLD LEGS	N/A	1,3
34	1LT-6 (LT-V2)	ILRT ROTOMETER (LOCKED CLOSED)	N/A	2,3
41	1SA-80 (SA-V14)	SERVICE AIR (LOCKED CLOSED)	N/A	2,3
42	1ED-119 (WL-D651)	RCDT PUMP DISCH BYPASS (LOCKED CLOSED)	N/A	2,3
44	1SF-145 (SF-D164)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
44	1SF-144 (SF-D165)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
45	1SF-118 (SF-D25)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
45	1SF-119 (SF-D26)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
39	1CC-250 (CC-V50)	CCW FROM RCP THERMAL BARRIER	N/A	N/A

MOVE THIS VALVE UNDER ITEM 11 FOR CHECK VALVES.

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
40	1RC-164 (RC-V525)	DEMIN WATER TO PRT	N/A	N/A
41	1SA-82 (SA-V15)	SERVICE AIR	N/A	N/A
59	CB-V1 (CB-V1)	CONTAINMENT VACUUM RELIEF	N/A	N/A
61	CM-V1 (CM-V1)	H_2 PURGE MAKEUP	N/A	N/A
76A	1SI-182 (SI-V150)	ACCUMULATORY FILL FROM RWST	N/A	N/A
77A	1SI-290 (SI-V188)	N_2 TO ACCUMULATORS	N/A	N/A
79	1FP-357 (FP-V48)	FIRE WATER STANDPIPE SUPPLY	N/A	N/A
80	1AI-220 (AI-V33)	INSTRUMENT AIR SUPPLY	N/A	N/A
90	1DW-65 (DW-V121)	DEMIN WATER SUPPLY	N/A	N/A
92	1SW-233 (SW-V142)	SERVICE WATER TO NNS FAN COILS	N/A	N/A
94A	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
94B	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
94C	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
95A	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
95B	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1

ALL [illegible]S ON THIS PAGE
SHOULD BE MOVED UNDER ITEM
11 FOR CHECK VALVES

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TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER			LIFT SETTING (± 1%)*	ORIFICE SIZE (IN.²)
STEAM GENERATOR				
A	B	C		
1MS-43	1MS-44	1MS-45	1170 psig	16.0
1MS-46	1MS-47	1MS-48	1185 psig	16.0
1MS-49	1MS-50	1MS-51	1200 psig	16.0
1MS-52	1MS-53	1MS-54	1215 psig	16.0
1MS-55	1MS-56	1MS-57	1230 psig	16.0

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency bus~~s~~es, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each motor-driven pump 1590 develops a discharge pressure of greater than or equal to ~~1510~~ psig at a recirculation flow of greater than or equal to 50 gpm.

2. Verifying that the steam turbine-driven 1510 pump develops a discharge pressure of greater than or equal to ~~1450~~ psig on a recirculation flow of greater than or equal to 90 gpm when the secondary steam supply pressure is greater than 210 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

SURVEILLANCE REQUIREMENTS (Continued)

Revisions 2, March 1978, and the system flow rate is 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980; and

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.1 inches water gauge while operating the system at a flow rate of 4000 cfm ± 10%;

2. Verifying that, on ~~a~~ EITHER A safety injection ~~and~~ OR A high radiation test signal, the system automatically switches into an isolation with recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 315 cfm relative to adjacent areas during system operation;

4. Verifying that the heaters dissipate 14 ± 1.4 kW when tested in accordance with ANSI N510-~~1975~~ 1980; and

5. Verifying that, on a High Chlorine test signal, the system automatically isolates the control room within 15 seconds and initiates a recirculation flow through the HEPA filters and charcoal adsorber banks.

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SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and ←

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%. ←

3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent RAB Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one RAB Emergency Exhaust System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each RAB Emergency Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6800 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980;

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,

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meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is less than 4.1 inches water gauge while operating the unit at a flow rate of 6800 cfm ± 10%,

2. Verifying that the system starts on a safety injection test signal,

3. Verifying that the system maintains the areas served by the exhaust system at a negative pressure of greater than or equal to 1/8 inch water gauge relative to the outside atmosphere,

4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-~~1975~~ 1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the unit at a flow rate of 6800 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6800 cfm ± 10%.

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3/4.7.10 ~~FIRE SUPPRESSION SYSTEMS~~ DELETED

~~FIRE PROTECTION WATER SUPPLY AND DISTRIBUTION SYSTEM~~

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Protection Water Supply and Distribution System shall be OPERABLE with:

a. At least two fire pumps, each with a capacity of 2100 gpm, with their discharges aligned to the fire suppression header,

b. The auxiliary reservoir water level shall be maintained in accordance with Specification 3.7.5, and

c. An OPERABLE flow path capable of taking suction from the auxiliary reservoir and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve on each spray system required to be OPERABLE per Specifications 3.7.10.2, 3.7.10.3, and 3.7.10.4.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Protection Water Supply and Distribution System otherwise inoperable, establish a backup system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The Fire Protection Water Supply and Distribution System shall be demonstrated OPERABLE:

a. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes on relief valve flow,

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

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SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each pump develops at least 2100 gpm at a discharge pressure of 131 psig, by testing at three points along the pump performance curve,

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that each fire suppression pump starts sequentially to maintain the Fire Suppression Water System pressure greater than or equal to 80 psig.

e. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.10.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The fuel storage tank contains at least 130 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes on relief valve flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D4057-81 is within the acceptable limits specified in Table 1 of ASTM D975-1981 when checked for viscosity and water and sediment, and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.10.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

PLANT SYSTEMS

FIRE SUPPRESSION SYSTEMS

FIRE PROTECTION WATER SUPPLY AND DISTRIBUTION SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months by verifying that:

   1. The batteries and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

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PLANT SYSTEMS

PREACTION AND MULTICYCLE SPRINKLER SYSTEMS

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LIMITING CONDITION FOR OPERATION

3.7.10.2 The Preaction and Multicycle Sprinkler Systems listed on Table 3.7-3 shall be OPERABLE:

APPLICABILITY: Whenever equipment protected by the Preaction and Multicycle Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Preaction and Multicycle Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Preaction and Multicycle Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a thermal test signal,

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   c) Performing a main drain test.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

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TABLE 3.7-3

~~PREACTION AND MULTICYCLE SPRINKLER SYSTEMS~~ DELETED

SYSTEM/ ZONE	DESCRIPTION	LOCATION/ ELEVATION
1/A	Airborne Radioactivity Removal Units - 1A & 1B Sprinkler (1-C-1-CHFA & 1-C-1-CHFB)	CNMT*/221
1/B	Containment Fan Coolers 1A-SA & 1B-SB Sprinkler (1-C-1-BAL)	CNMT*/236
1/C	Containment Fan Coolers 1A-SB & 1B-SB Sprinkler (1-C-1-BAL)	CNMT*/236
1/D	Pressurizer Cable Conduits (1-C-1-BAL)	CNMT*/236
1/E	Pressurizer Cable Trays (1-C-1-RCP-1B)	CNMT*/236
1/F	Electrical Cable Penetration Area - 1A Sprinkler (1-C-3-EPA)	CNMT*/261
1/G	Electrical Cable Penetration Area - 1B Sprinkler (1-C-3-EPB)	CNMT*/261
1/H	Conduit and Cable Trays - RC Pump 1B Area (1-C-1-RCP-1B)	CNMT*/261
1/I	Pressurizer Area (1-C-1-BAL)	CNMT*/286
2	Containment Spray and RHR Pump Room 1A Sprinkler (1-A-1-PA)	RAB/190
3	Containment Spray and RHR Pump Room 1B Sprinkler (1-A-1-PB)	RAB/190
4/A	Miscellaneous Pump and Equipment Room - South (1-A-2-MP)	RAB/216
4/B	Miscellaneous Pump and Equipment Room - North (1-A-2-MP)	RAB/216
5/A	Access Corridor Cable Trays (1-A-3-COR)	RAB/236
5/B	Mechanical Penetration Area (1-A-3-MP)	RAB/236
5/C 5/D	Aux. Feedwater Pumps and Component Cooling Water Heat Exchanger and Pumps Sprinkler (1-A-3-PB)	RAB/236
5/E	Decontamination Area and Corridor Cable Tray Sprinkler (1-A-3-COMB, 1-A-3-COME, 1-A-3-COMI)	RAB/236

*The sprinkler systems located within the containment building are not required to be operable during the performance of Type A containment leakage rate tests.

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~~TABLE 3.7-3 (Continued)~~

~~PREACTION AND MULTICYCLE SPRINKLER SYSTEMS~~

SYSTEM/ ZONE	DESCRIPTION	LOCATION/ ELEVATION
6/A	HVAC Chiller Equipment Area and Cable Tray Sprinkler (1-A-4-CHLR)	RAB/261
6/B	Corridor Cable Tray Sprinkler (1-A-4-COMB & 1-A-4-COME)	RAB/261
6/C	Charcoal Filter Room 1A & Corridor Cable Tray Sprinkler (1-A-4-COMI & 1-A-4-CHFA)	RAB/261
6/D	Charcoal Filter Room 1B Sprinkler (1-A-4-CHFB)	RAB/261
6/E	Electrical Penetration Area SA Sprinkler (1-A-EPA)	RAB/261
6/F	Electrical Penetration Area SB Sprinkler (1-A-EPB)	RAB/261
7/A	Cable Spreading Rooms A & B Sprinkler (1-A-CSRA & 1-A-CSRB)	RAB/286
7/B	HVAC Units E-17 & E-18 (12-A-5-CHF)	RAB/286
8/A	HVAC Equipment Room Sprinkler (12-A-6-HV7)	RAB/305
8/B	HVAC Units E-19 & E-20 (12-A-6-CHF-1)	RAB/305
9	Emergency Exhaust System E-12 & E-13 (5-F-3-CHFA & 5-F-3-CHFB)	FHB/261
10	Fuel Pool Cooling Heat Exchangers and Pumps (5-F-2-FPC)	FHB/236
11/A	Diesel Generator Room A 1A-Sprinkler (1-D-1-DGA-RM)	DGB/261
11/B	Diesel Generator Fuel Oil Day Tank A Enclosure 1A-Sprinkler (1-D-DTA)	DGB/261/280
12/A	Diesel Generator Room B 1B-Sprinkler (1-D-1-DGB-RM)	DGB/261
12/B	Diesel Generator Fuel Oil Day Tank B Enclosure 1B-Sprinkler (1-D-DTB)	DGB/261/280
13/A	Diesel Oil Pump Room A 1A-Sprinkler (1-O-PA)	DFOSB/242.25
13/B	Diesel Oil Pump Room B 1B-Sprinkler (1-O-PB)	DFOSB/242.25

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The fire hose stations given in Table 3.7-4 shall be OPERABLE.*

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-4 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION requirement shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.3 Each of the fire hose stations given in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months, by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,
   2. Removing the hose for inspection and re-racking, and
   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and
   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

*Fire hose stations within the containment are required to be operable only during refueling and maintenance outages.

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TABLE 3.7-4

~~FIRE HOSE STATIONS~~ DELETED

LOCATION[1]	ELEVATION	HOSE RACK NO.
CNMT	221	221-C-4
CNMT	221	221-C-12
CNMT	221	221-C-19
CNMT	236	236-C-4
CNMT	236	236-C-12
CNMT	236	236-C-19
CNMT	261	261-C-4
CNMT	261	261-C-12
CNMT	261	261-C-19
CNMT	286	286-C-4
CNMT	286	286-C-12
CNMT	286	286-C-19
RAB	190	190-G-16
RAB	190	190-G-38
RAB	216	216-G-16
RAB	216	216-Fz-27
RAB	216	216-G-38
RAB	216	216-Gy-13
RAB	236	236-Gy-13
RAB	236	236-G-16
RAB	236	236-Fz-27
RAB	236	236-D-27
RAB	236	236-G-38
RAB	236	236-Kz-31
RAB	236	236-C-39
RAB	236	236-Fw-43
RAB	236	236-Jz-43
RAB	236	236-E-15
RAB	261	261-Gy-13
RAB	261	261-E-15
RAB	261	261-G-16
RAB	261	261-D-27
RAB	261	261-Kz-31
RAB	261	261-G-38
RAB	261	261-C-39
RAB	261	261-Fw-42
RAB	286	286-C-15
RAB	286	286-E-15
RAB	286	286-G-16
RAB	286	286-E-38
RAB	286	286-C-39
RAB	286	286-Jv-41
RAB	286	286-Fw-42

[1]CNMT - Containment Building FHB - Fuel Handling Building
RAB - Reactor Auxiliary Building DGB - Diesel Generator Building

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TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	MAXIMUM TEMPERATURE LIMIT (°F)
REACTOR AUXILIARY BUILDING	
1. Control Room Envelope, (El 305')	85
2. Process I&C, Room (El 305')	85
3. Rod Control Cabinets Area (El 305')	104
4. A&B Battery Rooms (El 286')	85
5. A&B Switchgear Rooms (El 286')	90
6. Main Steam, Feedwater Pipe Tunnel (El 286' & 261')	116
7. SA&SB Electrical Penetration Areas (El ~~236')~~ 261 & 286	104
8. Area with MCC 1A35SA and 1B35SB (El 261')	104
9. HVAC Chillers, Auxiliary FW Piping & Valve Area (El ~~236')~~ 261	104
10. CCW Pumps, CCW Hx, Auxiliary FW Pumps Area (El 236')	104
11. 1A-SA, 1B-SB, and 1C-SAB ~~Spare~~ Charging Pump Rooms (El 236')	104
12. Service Water Booster Pump 1B-SB (El 236')	104
13. Mechanical and Electrical Penetration Areas (El 236')	104
14. Containment Spray Additive Tank, and H&V Equipment Area (El 216')	104
15. Trains A&B Containment Spray Pump, RHR Pump, H&V Equipment Areas (El 190')	104
FUEL HANDLING BUILDING	
16. Trains A&B Emergency Exhaust System Areas (El 261')	104
17. Fuel Pool Cooling Pump and Heat Exchanger Area (El 236')	104
WASTE PROCESSING BUILDING	
18. H&V Equipment Room (El 236')	104
MISCELLANEOUS	
19. ~~Condensate Storage~~ Tank Area (El ~~261~~ 236')	122
20. Diesel Fuel Oil Storage Building (El 242')	122
21. Emergency Service Water Electrical Equipment Room	104
22. Emergency Service Water Pump Room	122
23. ~~1A-SA & 1B-SB Exhaust Silencer Rooms (El 292')~~	~~122~~ DELETED
24. 1A-SA & 1B-SB H&V Equipment Rooms (El 292')	122
25. 1A-SA & 1B-SB H&V Equipment Rooms (El 280')	~~110~~ 118
26. 1A-SA & 1B-SB Electrical Rooms (El 261')	~~104~~ 116
27. 1A-SA & 1B-SB Diesel Generator Rooms (El 261')	120

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 2670 gallons of fuel, which is equivalent to ~~92.5~~ 85% indicated level,

2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

3. A separate fuel oil transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either emergency diesel generator (EDG) has not been successfully tested within the 24 hours preceding entry into this ACTION, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 separately for each such EDG within 24 hours. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 within 24 hours*#; restore the

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*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#~~The diesel shall not be rendered inoperable by activities performed to support testing pursuant to this ACTION statement (e.g., an air roll).~~
ACTIVITIES WHICH NORMALLY SUPPORT TESTING PURSUANT TO 4.8.1.1.2.a.4 WHICH WOULD RENDER THE DIESEL INOPERABLE (eg AIR ROLL) SHALL NOT BE PERFORMED FOR TESTING REQUIRED BY THIS ACTION STATEMENT

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

diesel generator to OPERABLE status within 72 hours or be IN at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See also ACTION statement d. below.

c. With one offsite circuit of 3.8.1.1.a and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 within 8 hours*#; restore one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See also ACTION statement d. below. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a or b, as appropriate with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this Action Statement for an OPERABLE diesel or a restored to OPERABLE diesel satisfies the EDG test requirement of Action Statement a or b.

d. With one diesel generator inoperable, in addition to Action b and c above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE. If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, except as provided for in Action Statement d.2 below.

   2. If in MODES 1, 2, or 3 and the result of the inoperable diesel generator is that three auxiliary feedwater pumps are inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#~~The diesel shall not be rendered inoperable by activities performed to support testing pursuant to this ACTION statement (e.g., an air roll).~~ ACTIVITIES WHICH NORMALLY SUPPORT TESTING PURSUANT TO 4.8.1.1.2.a.4 WHICH WOULD RENDER THE DIESEL INOPERABLE (eg AIR ROLL) SHALL NOT BE PERFORMED FOR TESTING REQUIRED BY THIS ACTION STATEMENT.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued):

e. With two of the required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by sequentially performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 on both diesels within 8 hours#, unless the diesel generators are already operating; restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow Action Statement a with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test(s) of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this Action Statement for the OPERABLE diesels satisfies the EDG test requirement of Action Statement a.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow Action Statement b with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this Action Statement for a restored-to-OPERABLE diesel satisfies the EDG test requirement of Action Statement b.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignment and power availability, and

b. Demonstrated OPERABLE at least once per 18 months by manually transferring the onsite Class 1E power supply from the unit auxiliary transformer to the startup auxiliary transformer.

ACTIVITIES WHICH NORMALLY SUPPORT TESTING PURSUANT TO 4.8.1.1.2.a.4 WHICH WOULD RENDER THE DIESEL INOPERABLE (eg AIR ROLL) SHALL NOT BE PERFORMED FOR TESTING REQUIRED BY THIS ACTION STATEMENT.

~~#The diesel shall not be rendered inoperable by activities performed to support testing pursuant to this ACTION statement (e.g., an air roll).~~

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 26 degrees but less than or equal to 38 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, IF ~~if~~ the gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tank, in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A.

e. At least once per 184 days, on a STAGGERED TEST BASIS, the diesel generators shall be started** and accelerated to at least 450 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds after the start signal.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

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SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected shutdown loads through the load sequencer, and operating for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz.

5. Verifying that on a safety injection test signal (without loss of power) the diesel generator starts** on the auto-start signal and operates on standby for greater than or equal to 5 minutes.

6. Simulating a loss of offsite power in conjunction with a safety injection test signal, and

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected emergency (accident) loads through the sequencing timers, and operating for greater than or equal to 5 minutes and maintaining the steady-state voltage and frequency at 6900 ± 690 volts and 60 ± 1.2 Hz.

c) Verifying that all diesel generator trips, except engine overspeed, LOSS OF GENERATOR POTENTIAL TRANSFORMER CIRCUIT generator differential, and emergency bus differential are automatically bypassed upon loss of offsite power signal IN CONJUNCTION WITH a safety injection signal.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

SURVEILLANCE REQUIREMENTS (Continued)

7. Verifying the diesel generator operates** for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 6800-7000 kW*** and, during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6200-6400 kW***. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.f.6 b).#

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 6500 kW;

9. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Proceed through its shutdown sequence.

10. Verifying that the following diesel generator lockout features prevent diesel generator operation:

    a) Engine overspeed
    b) Generator differential
    c) Emergency bus differential
    d) Emergency Stop
    e) Operational and maintenance switch in the maintenance mode.
    f) LOSS OF GENERATOR POTENTIAL TRANSFORMER CIRCUIT

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 450 rpm in less than or equal to 10 seconds.

h. At least once per 10 years by:

   1) Draining each main fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or other appropriate cleaning solution, and

---

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

#If Specification 4.8.1.1.2f.6 b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 6200-6400 kW for 1 hour or until operating temperature has stabilized.

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ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day tank containing a minimum volume of 2670 gallons of fuel, which is equivalent to ~~92.5~~ 85% indicated level,

2. A separate main fuel oil storage tank containing a minimum volume of 100,000 gallons of fuel, and

3. A fuel oil transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over irradiated fuel and within 8 hours, depressurize and vent the Reactor Coolant System through a vent of greater than or equal to 2.9 square inches. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1 and 4.8.1.1.2.

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LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one ~~of the~~ 118-volt A.C. vital bus NOT ENERGIZED FROM ITS associated inverter ~~is not~~ connected to its associated D.C. bus, ~~reconnect the inverter~~ REENERGIZE THE 118-VOLT A.C. VITAL BUS THROUGH ITS ASSOCIATED INVERTER CONNECTED to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) given in Table 3.8-1 inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be demonstrated OPERABLE:*

a. At least once per 18 months:

1. By verifying that the 6900-volt circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of

* FUSES ARE PROVIDED FOR TABLE COMPLETENESS ONLY; THERE ARE NO SURVEILLANCE REQUIREMENTS.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
29	Reactor Coolant Pump (1A-SN)	Relay Trips Feeder Breaker	Relay Trips Upstream Breaker
30	Lighting Panel (LP-105)	~~70~~ 150 A Breaker	~~150~~ 70 A Breaker
31	Lighting Panel (LP-106)	~~50~~ 100 A Breaker	~~60~~ 50 A Breaker
32	Lighting Panel (LP-101)	~~60~~ 125 A Breaker	~~125~~ 60 A Breaker
33	Lighting Panel (LP-102)	~~60~~ 125 A Breaker	~~125~~ 60 A Breaker
34	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
35	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
36	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
37	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
38	Elevator Disc Switch	100 A Breaker	100 A Breaker
39	Power Receptacles 1-2 & 1-6	60 A Breaker	60 A Breaker
40	Power Receptacles 1-9 & 1-13	60 A Breaker	60 A Breaker
41	Power Receptacles 1-10 & 1-14	60 A Breaker	60 A Breaker
42	Reactor Coolant Pump 1A-SN Oil BRG Lift Pump	30 A Breaker	30 A Breaker
43	Disk Switch for 5-Ton Monorail	50 A Breaker	50 A Breaker
44	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
45	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
62	IRVH Cable Bridge Hoist	15 A Breaker	15 A Breaker
63	AOV-1RC-P525SN-1	6 A Fuse	20 A Breaker
64	MOV-2SI-V537SA-1 (8808A) Pos. SW. ANN	3 A Fuse	15 A Breaker
65	Integrated Head Cooling Fan E-80(1A-NNS)	20 A Breaker	20 A Breaker
66	Integrated Head Cooling Fan E-81 (1A-NNS)	20 A Breaker	20 A Breaker
67	AOV-2BD-F6SN-1 (PCV-8400A)	6 A Fuse	15 A Breaker
68	Damper (CV-D9-1)	6 A Fuse	15 A Breaker
69	Damper (CV-D13-1)	6 A Fuse	15 A Breaker
70	Con. Rod Drive Mech. Fan E-80 (1A-NNS)	6 A Fuse	15 A Breaker
71	Con. Rod Drive Mech. Fan E-81 (1A-NNS)	6 A Fuse	15 A Breaker
72	Reactor Coolant Pump (1A-SN) Space Heater	15 A Breaker	30 A Breaker
73	Inst. Rack C1-R1	20 A Breaker	20 A Breaker
74	AH-37 (1A-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
75	AH-38 (1A-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
76	AH-39 (1A-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
77	Elevator Equipment Room Fan (E-3) (1X-NNS)	20 A Breaker	20 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
78	SV-7SP ~~SV7-SP~~-V334-1 (Rad. Mon. Sampling Valves)	15 A Breaker	15 A Breaker
79	SV-7SP ~~SV7-SP~~-V318-1	15 A Breaker	15 A Breaker
80	SV-7SP ~~SV7-SP~~-V320-1	15 A Breaker	15 A Breaker
81	Containment Atmo. Rad. Mon. Valve (7SP-V322-1)	15 A Breaker	15 A Breaker
82	Containment Atmo. Rad. Mon. Valve (7SP-V324-1)	15 A Breaker	15 A Breaker
83	Containment Atmo. Rad. Mon. Valve (7SP-V326-1)	15 A Breaker	15 A Breaker
84	Containment Atmo. Rad. Mon. Valve (7SP-V328-1)	15 A Breaker	15 A Breaker
85	Containment Atmo. Rad. Mon. Valve (7SP-330-1)	15 A Breaker	15 A Breaker
86	Containment Atmo. Rad. Mon. Valve (7SP-332-1)	15 A Breaker	15 A Breaker
87	AOV-2RC ~~AOV2RC~~-D528SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
88	MOV-2CS-V516SA-1 (8112) (Limit Switch)	8 A Fuse	15 A Breaker
89	AOV-2CS-V511SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
90	AOV-2CS-V512SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
91	AOV-2CS-V513SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
92	MOV-2SI-V537SA-1 (8808A) (Limit Switch)	8 A Fuse	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
122	Cont. Fan Cooler AH-2 (1A-SA) Space Heater	15 A Breaker	15 A Breaker
123	Cont. Fan Cooler AH-2 (1B-SA) Space Heater	15 A Breaker	15 A Breaker
124	Cont. Fan Cooler AH-3 (1A-SA) Space Heater	15 A Breaker	15 A Breaker
125	Cont. Fan Cooler AH-3 (1B-SA) Space Heater	15 A Breaker	15 A Breaker
126	Primary Shield Cooling Fan S-2 (1A-SA) Heater	15 A Breaker	15 A Breaker
127	Hydrogen Recombiner	125 A Breaker	125 A Breaker
128	Reactor Support Cooling Fan S-4 (1A-SA)	100 A Breaker	100 A Breaker
129	MOV-1RH-V501SA-1 (8701B) (Isolation Valve)	15 A Breaker	15 A Breaker
130	MOV-2SI-V537SA-1 (8808A) (Accumulator "A" Discharge Valve)	40 A Breaker	40 A Breaker
131	MOV-2SI-V535SA-1 (8808C) (Accumulator "C" Discharge Valve)	40 A Breaker	40 A Breaker
132	MOV-2CS-V516SA-1 (8112) ~~(8812)~~ (RCP Seal Water Return Isolation Valve)	15 A Breaker	15 A Breaker
133	MOV-1RH-V503SA-1 (8701A) (RHRS Inlet Isolation Valve)	15 A Breaker	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
237	MOV-1RC-V528SN-1 (8000C)	15 A Breaker	15 A Breaker
238	Lighting Panel LP-104	~~70~~ 150 A Breaker	~~150~~ 70 A Breaker
239	Lighting Panel LP-107. (N/E)	~~50~~ 100 A Breaker	50 A Breaker
240	Lighting Panel LP-103	~~50~~ 100 A Breaker	~~100~~ 50 A Breaker
241	Lighting Panel LP-123	~~60~~ 125 A Breaker	~~125~~ 60 A Breaker
242	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
243	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
244	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
245	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
246	Power Receptacles #1-12, 1-16	60 A Breaker	60 A Breaker
247	Power Receptacles #1-3, 1-7	60 A Breaker	60 A Breaker
248	Power Receptacles #1-4, 1-8	60 A Breaker	60 A Breaker
249	RCP-1B-SN Oil Bearing Lift Pump	30 A Breaker	30 A Breaker
250	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
251	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
252	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
267	Charcoal Temp. Detection Fan S-1 (1A-NNS)	6 A Fuse	15 A Breaker
268	Airborne Radioactivity Removal Unit S-1 (1A-NNS)	90 A Breaker	90 A Breaker
269	RCP-1C-SN Oil Bearing Lift Pump	30 A Breaker	30 A Breaker
270	Containment Building Sump Pump 1A-NNS	50 A Breaker	50 A Breaker
271	Airborne Radioactivity Removal Unit S-1 (1B-NNS)	90 A Breaker	90 A Breaker
272	Fuel Transfer Cont. Cab (Pump Motor)	15 A Breaker	15 A Breaker
273	RCC Change Fixt (Gripper Hoist Ratio Motor)	15 A Breaker	15 A Breaker
274	Fuel Transfer Manipulator Crane	30 A Breaker	30 A Breaker
275	RA-ICA-3584	0.6 A FUSE	20 A BREAKER
276	RA-ICA-3585	0.6 A FUSE	15 A BREAKER
277	RA-ICA-3586	0.6 A FUSE	15 A BREAKER
278	RA-ICA-3587	0.6 A FUSE	15 A BREAKER
279	2SI-V537-SA-1 LIMIT SWITCH	8 A FUSE	15 A BREAKER
280	2MD-V36 SA-1 LIMIT SWITCH	8 A FUSE	15 A BREAKER
281	DAMPER CV-D3 SA-1	6 A FUSE	20 A BREAKER
282	AOV-2CP-B1 SA-1	6 A FUSE	20 A BREAKER
283	DAMPER CV-D5 SA-1	6 A FUSE	20 A BREAKER
284	RA-ICA-3575	0.6 A FUSE	15 A BREAKER

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
285	RA-ICA-3576	0.6 A FUSE	15 A BREAKER
286	RA-ICA-3577	0.6 A FUSE	20 A BREAKER
287	RA-ICA-3582	0.6 A FUSE	20 A BREAKER
288	RA-ICA-3583	0.6 A FUSE	20 A BREAKER
289	2BD-F4SN-1 (PCV-8400B) POSITION SWITCHES	6 A FUSE	15 A BREAKER
290	2BD-F5SN-1 (PCV-8400C) POSITION SWITCHES	6 A FUSE	15 A BREAKER
291	MOV-2SI-535 SA-1 (8808C) ANN & POSITION SWITCHES	3 A FUSE	15 A BREAKER
292	LOCAL FIS-1AR-7647A FAN S1 (1A-NNS) FLOW SWITCH	6 A FUSE	15 A BREAKER
293	LOCAL FS-7647B FAN S1 (1A-NNS) FLOW SWITCH	6 A FUSE	15 A BREAKER
294	DAMPER AR-D4-1 LIMIT SWITCH	6 A FUSE	15 A BREAKER
295	FIRE DETECTION CONTROL PANEL FAN S1 (1B-NNS)	6 A FUSE	15 A BREAKER
296	FUEL TRANSFER CONSOLE REACTOR SIDE 120/208 V SUPPLY	20 A BREAKER	20 A BREAKER
297	FIRE DETECTION CONTROL PANEL FAN S1 (1B-NSS)	20 A BREAKER	20 A BREAKER
298	CONTAINMENT FAN COOLER AH-1 (1A-SB)	225 A BREAKER	1600 A BREAKER
299	CONTAINMENT FAN COOLER AH-1 (1A-SB)	225 A BREAKER	1600 A BREAKER
300	CONTAINMENT FAN COOLER AH-1 (1B-SB)	225 A BREAKER	1600 A BREAKER
301	CONTAINMENT FAN COOLER AH-1 (1B-SB)	225 A BREAKER	1600 A BREAKER

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LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve given in Table 3.8-2 shall be bypassed only under accident conditions by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not capable of being bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The thermal overload protection for the above required valves shall be verified to be bypassed only under accident conditions by an OPERABLE integral bypass device by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST* of the bypass circuitry:

a. At least once per 92 days for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions; and

b. Following maintenance on the motor starter.

* THIS TEST SHALL COVER THE BYPASS CIRCUITRY FROM THE MASTER BYPASS RELAY IN THE SEQUENCER THROUGH OPERATION OF THE LOCAL BYPASS RELAY.

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TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1CS-341 (2CS-V522)	RCP A SEAL ISOL	YES
1CS-382 (2CS-V523)	RCP B SEAL ISOL	YES
1CS-423 (2CS-V524)	RCP C SEAL ISOL	YES
1CS-182 (2CS-V600)	CSIP A MINIFLOW ISOLATION	YES
1CS-210 (2CS-V601)	CSIP B MINIFLOW ISOLATION	YES
1CS-196 (2CS-V602)	CSIP C MINIFLOW ISOLATION	YES
1CS-235 (2CS-V609)	CSIP to RCS ISOLATION	YES
1CS-166 (2CS-L521)	VCT ISOLATION	YES
1CS-292 (2CS-L522)	RWST ISOLATION	YES
1CS-214 (2CS-V585)	CSIPS MINIFLOW ISOLATION	YES
1CS-165 (2CS-L520)	VCT ISOLATION	YES
1CS-291 (2CS-L523)	RWST ISOLATION	YES
1CS-238 (2CS-V610)	CSIP TO RCS ISOLATION	YES
1CS-170 (2CS-V587)	CSIP SUCTION ISOLATION	YES
1CS-169 (2CS-V589)	CSIP SUCTION ISOLATION	YES
1CS-171 (2CS-V590)	CSIP SUCTION ISOLATION	YES
1CS-168 (2CS-V588)	CSIP SUCTION ISOLATION	YES
1CS-219 (2CS-V603)	CSIP DISCHARGE ISOL	YES
1CS-217 (2CS-V604)	CSIP DISCHARGE ISOL	YES
1CS-218 (2CS-V605)	CSIP DISCHARGE ISOL	YES
1CS-220 (2CS-V606)	CSIP DISCHARGE ISOL	YES
1CS-240 (2CS-V611)	SEAL WATER INJECTION	YES
1CS-278 (2CS-V586)	BORIC ACID TANK TO CSIP	YES
1CS-746 (2CS-V757)	CSIP MINIFLOW	YES
1CS-752 (2CS-V759)	CSIP MINIFLOW	YES
1CS-753 (2CS-V760)	CSIP MINIFLOW	YES
1CS-745 (2CS-V758)	CSIP MINIFLOW	YES
1CS-472 (2CS-V517)	RCP~~S~~ SEAL WATER RETURN ISOL	YES
1CS-470 (2CS-V516)	RCP SEAL WATER ISOLATION	YES
1RH-25 (2RH-V507)	RHR TO CSIP SUCTION	YES
1RH-63 (2RH-V506)	RHR TO CSIP SUCTION	YES
1RH-31 (2RH-F513)	RHR A MINI FLOW	YES
1RH-69 (2RH-F512)	RHR B MINI FLOW	YES
1RH-2 (1RH-V503)	RHRS INLET ISOLATION	YES
1RH-40 (1RH-V501)	RHRS INLET ISOLATION	YES
1RH-1 (1RH-V502)	RHRS INLET ISOLATION	YES
1RH-39 (1RH-V500)	RHRS INLET ISOLATION	YES
1SI-1 (2SI-V503)	BORON INJECTION TANK INLET ISOL	YES
1SI-4 (2SI-V506)	BORON INJECTION TANK OUTLET ISOL	YES
1SI-2 (2SI-V504)	BORON INJECTION TANK INLET ISOL	YES
1SI-3 (2SI-V505)	BORON INJECTION TANK OUTLET ISOL	YES
1SI-246 (2SI-V537)	ACCUMULATOR A DISCHARGE ISOLATION	YES
1SI-248 (2SI-V535)	ACCUMULATOR C DISCHARGE ISOLATION	YES
1SI-300 (2SI-V571)	CNMT SUMP TO RHR PUMP A ISOL	YES
1SI-310 (2SI-V573)	CNMT SUMP TO RHR PUMP A ISOL	YES
1SI-247 (2SI-V536)	ACCUM B DISCHARGE ISOLATION	YES

TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1SI-301 (2SI-V570)	CNMT SUMP TO RHR PUMP B ISOL	YES
1SI-311 (2SI-V572)	CNMT SUMP TO RHR PUMP B ISOL	YES
1SI-107 (2SI-V500)	HH SI TO RCS HL	YES
1SI-52 (2SI-V502)	HH SI TO RCS CL	YES
1SI-86 (2SI-V501)	HH SI TO RCS HL	YES
1SI-326 (2SI-V577)	LH SI TO RCS HL	YES
1SI-327 (2SI-V576)	LH SI TO RCS HL	YES
1SI-340 (2SI-V579)	LH SI TO RCS CL	YES
1SI-341 (2SI-V578)	LH SI TO RCS CL	YES
1SI-359 (2SI-V587)	LH SI TO RCS HL	YES
1SI-322 (2SI-V575)	RWST TO RHR A ISOL	YES
1SI-323 (2SI-V574)	RWST TO RHR B ISOL	YES
1CC-128 (3CC-B5)	CCS NONESSENTIAL RETURN ISOL	YES
1CC-127 (3CC-B6)	CCS NONESSENTIAL RETURN ISOL	YES
1CC-99 (3CC-B19)	CCS NONESSENTIAL RETURN ISOL	YES
1CC-113 (3CC-B20)	CCS NONESSENTIAL RETURN ISOL	YES
1CC-147 (3CC-V165)	RHR COOLING ISOL	YES
1CC-167 (3CC-V167)	RHR COOLING ISOL	YES
1CC-176 (2CC-V172)	CVCS HX CNMT ISOLATION	YES
1CC-202 (2CC-V182)	CVCS HX CNMT ISOLATION	YES
1CC-208 (2CC-V170)	CCW-RCPS ISOLATION	YES
1CC-299 (2CC-V183)	RCPS BEARING HX ISOLATION	YES
1CC-251 (2CC-V190)	RCPS THER BARRIER ISOLATION	YES
1CC-207 (2CC-V169)	CCW-RCPS ISOLATION	YES
1CC-297 (2CC-V184)	RCPS BEARING HX ISOLATION	YES
1CC-249 (2CC-V191)	RCPS THER BARRIER ISOLATION	YES
1CT-105 (2CT-V6)	CNMT SPRAY SUMP A RECIRC ISOL	YES
1CT-102 (2CT-V7)	CNMT SPRAY SUMP B RECIRC ISOL	YES
1CT-26 (2CT-V2)	CNMT SPRAY PUMP A INJECT. SUPPLY	YES
1CT-71 (2CT-V3)	CNMT SPRAY PUMP B INJECT. SUPPLY	YES
1CT-50 (2CT-V21)	SPRAY HDR A ISOLATION	YES
1CT-12 (3CT-V85)	NAOH ADDITIVE ISOLATION	YES
ICT-88 (2CT-V43)	SPRAY HDR B ISOLATION	YES
ICT-11 (3CT-V88)	NAOH ADDITIVE ISOLATION	YES
1CT-47 (2CT-V25)	CNMT SPRAY HDR A RECIRC	YES
1CT-24 (2CT-V8)	CNMT SPRAY PUMP A EDUCTOR TEST	YES
1CT-95 (2CT-V49)	CNMT SPRAY HDR B RECIRC	YES
1CT-25 (2CT-V145)	CNMT SPRAY PUMP B EDUCTOR TEST	YES
1AF-5 (3AF-V187)	AFWP A RECIRC	YES
1AF-24 (3AF-V188)	AFWP B RECIRC	YES
1AF-55 (2AF-V10)	AFW TO SG A ISOL	~~YES~~ NO*
1AF-93 (2AF-V19)	AFW TO SG B ISOL	~~YES~~ NO*
1AF-74 (2AF-V23)	AFW TO SG C ISOL	~~YES~~ NO*
1AF-137 (2AF-V116)	AFWTD TO SG A ISOL	~~YES~~ NO*
1AF-143 (2AF-V117)	AFWTD TO SG B ISOL	~~YES~~ NO*
1AF-149 (2AF-V118)	AFWTD TO SG C ISOL	~~YES~~ NO*
1MS-70 (2MS-V8)	AFWTD STEAM B ISOLATION	~~YES~~ NO*

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1MS-72 (2MS-V9)	AFWTD STEAM C ISOLATION	~~YES~~ NO*
1SW-39 (3SW-B5)	NORMAL SW HDR A ISOLATION	YES
1SW-276 (3SW-B8)	NORMAL SW HDR A RETURN ISOL	YES
1SW-270 (3SW-B15)	SW HDR A TO AUX RSVR ISOL	YES
1SW-40 (3SW-B6)	NORMAL SW HDR B ISOL	YES
1SW-275 (3SW-B13)	SW HDR A RETURN ISOL	YES
1SW-274 (3SW-B14)	SW HDR B RETURN ISOL	YES
1SW-271 (3SW-B16)	SW HDR B TO AUX RSVR ISOL	YES
1SW-3 (3SW-B3)	EMER SW PUMP 1A MAIN RSVR INLET	YES
1SW-4 (3SW-B4)	EMER SW PUMP 1B MAIN RSVR INLET	YES
1SW-1 (3SW-B1)	EMER SW PUMP 1A AUX RSVR INLET	YES
1SW-2 (3SW-B2)	EMER SW PUMP 1B AUX RSVR INLET	YES
1SW-92 (2SW-B46)	SW TO FAN CLR AH3 INLET	YES
1SW-97 (2SW-B47)	SW ~~TO~~ FROM FAN CLR AH3 OUTLET	YES
1SW-91 (2SW-B45)	SW TO FAN CLR AH2 INLET	YES
1SW-109 (2SW-B49)	SW ~~TO~~ FROM FAN CLR AH2 OUTLET	YES
1SW-225 (2SW-B52)	SW TO FAN CLR AH1 INLET	YES
1SW-98 (2SW-B48)	SW ~~TO~~ FROM FAN CLR AH1 OUTLET	YES
1SW-227 (2SW-B51)	SW TO FAN CLR AH4 INLET	YES
1SW-110 (2SW-B50)	SW ~~TO~~ FROM FAN CLR AH4 OUTLET	YES
1SW-124 (3SW-B70)	SW TO AFWTD PUMP	YES
1SW-126 (3SW-B71)	SW TO AFWTD PUMP	YES
1SW-129 (3SW-B73)	SW TO AFWTD PUMP	YES
1SW-127 (3SW-B72)	SW TO AFWTD PUMP	YES
1SW-123 (3SW-B75)	SW TO AFW PUMP A SUPPLY	YES
1SW-121 (3SW-B74)	SW TO AFW PUMP A SUPPLY	YES
1SW-132 (3SW-B77)	SW TO AFW PUMP B SUPPLY	YES
1SW-130 (3SW-B76)	SW TO AFW PUMP B SUPPLY	YES
1ED-94 (2MD-V36)	CNMT SUMP ISOLATION	YES
1ED-95 (2MD-V77)	CNMT SUMP ISOLATION	YES
3CZ-B5	RAB ELEC PROT INLET	YES
3CZ-B6	RAB ELEC PROT INLET	YES
3CZ-B7	RAB ELEC PROT EXHAUST	YES
3CZ-B8	RAB ELEC PROT EXHAUST	YES
3CZ-B32	RAB ELEC PROT PURGE MAKE-UP	YES
3CZ-B33	RAB ELEC PROT PURGE MAKE-UP	YES
3CZ-B34	RAB ELEC PROT PURGE INLET	YES
3CZ-B35	RAB ELEC PROT PURGE INLET	YES
3FV-B2	FUEL HANDLING EXHAUST INLET	NO*
3FV-B4	FUEL HANDLING EXHAUST INLET	NO*
3CZ-B1	CONTROL ROOM NORMAL SUPPLY ISOL	NO*
3CZ-B3	CONTROL ROOM NORMAL EXHAUST ISOL	NO*
3CZ-B17	CONTROL ROOM PURGE MAKE UP	NO*
3CZ-B2	CONTROL ROOM NORMAL SUPPLY ISOL	NO*
3CZ-B4	CONTROL ROOM EXHAUST ISOLATION	NO*
3CZ-B18	CONTROL ROOM PURGE MAKE UP	NO*
3CZ-B14	CONTROL ROOM PURGE EXHAUST	NO*

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TABLE 4.9-1

ADMINISTRATIVE CONTROLS
TO PREVENT DILUTION DURING REFUELING

VALVE ~~LOCATION~~/ID	VALVE POSITION DURING REFUELING	LOCK	DESCRIPTION
1CS-149 (CS-D121 SN)	Closed	Yes	RMW to the CVCS makeup control system
1CS-510 (CS-D631 SN)	Closed	Yes	Boric Acid Batch Tank Outlet valve, may be opened if the batching tank concentration is $\geq$ 2000 ppm boron, and valve 1CS-503 (makeup water supply to batch tank) is closed.
1CS-503 (CS-D251)	Closed	Yes	RMW to Batching Tank. Do not open unless outlet valve 1CS-510 is closed.
1CS-570 (CS-D575 SN)	Closed	No	CVCS LETDOWN TO BTRS. Place valve in "shut" at valve control switch and place BTRS function selector switch in "off." No lock required.
1CS-670 (CS-D599 SN)	Closed	Yes	RMW to BTRS loop.
1CS-649 (CS-D198 SN)	Closed	Yes	Resin sluice to BTRS demineralizers.
1CS-93 (CS-D51 SN)	Closed	Yes	Resin sluice to CVCS demineralizers
1CS-320 (CS-D641 SN)	Closed	Yes	Recycle Evaporation Feed Pump to charging/safety injection pump suction.
1CS-98 (CS-D740 SN)	Open	No	BTRS bypass valve. Place valve control switch in "open" position.

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3/4.9.6 REFUELING MACHINE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine, used for movement of fuel assemblies, having:

1. A minimum capacity of 4000 pounds, and

2. An automatic overload cutoff limit less than or equal to 2700 pounds.

b. The auxiliary hoist, used for latching and unlatching drive rods, having:

1. A minimum capacity of 3000 pounds, and

2. A 1000-pound load indicator that shall be used to monitor loads to prevent lifting more than 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for the refueling machine and/or auxiliary hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE, within 100 hours prior to the start of such operations, by performing a load test of at least 4000 pounds and demonstrating an automatic load cutoff ~~when the refueling machine load exceeds~~ AT LESS THAN OR EQUAL TO 2700 pounds.

4.9.6.2 The auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 900 pounds.

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3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Emergency Exhaust System Trains shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in a storage pool.

ACTION:

a. With one Fuel Handling Building Emergency Exhaust System Train inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Handling Building Emergency Exhaust System Train is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorber.

b. With no Fuel Handling Building Emergency Exhaust System Trains OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Handling Building Emergency Exhaust System Train is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Emergency Exhaust System trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6600 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980.

4.9.12 (Continued)

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is not greater than 4.1 inches water gauge while operating the unit at a flow rate of 6600 cfm ± 10%,

2. Verifying that, on a High Radiation test signal, the system automatically starts ~~(unless already operating)~~ and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch water gauge, relative to the outside atmosphere, during system operation at a flow rate of 6600 cfm ± 10%,

4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-~~1975~~ 1980.

e: After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the unit at a flow rate of 6600 cfm ± 10%.

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SURVEILLANCE REQUIREMENTS (Continued)

4.9.12 (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6600 cfm ± 10%.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge or Vent	P Each PURGE(3) Grab Sample	P Each PURGE(3)	Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. a. Plant Vent Stack	M(3),(4),(5) Grab Sample		Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
b. Turbine Bldg Vent Stack; Waste Processing Bldg Vent Stacks 5&5A	M Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
4. All Release Types as listed in 1., 2., and 3. above	Continuous(6)	W(7) Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous(6)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(6)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(6)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -42 pcm/°F. The MTC value of -33 pcm/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -42 pcm/°F.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging/safety injection pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above ~~200~~ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1770 pcm after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16800 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 436,000 gallons of ~~2000 ppm~~ 2000-2200 ppm borated water be maintained in the refueling water storage tank (RWST).

With the RCS temperature below ~~200~~ 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 335°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1000 pcm after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4900 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 82,000 gallons of ~~2000~~ 2000-2200 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified value, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values. The specified percent level and gallons differ by less than 0.1%.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. THE RWST TEMPERATURE WAS SELECTED TO BE CONSISTENT WITH ANALYTICAL ASSUMPTIONS FOR CONTAINMENT HEAT LOAD.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

REMOTE SHUTDOWN SYSTEM (Continued)

This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 ~~FIRE DETECTION INSTRUMENTATION~~ DELETED

~~The OPERABILITY of the fire detection instrumentation ensures that both adequate~~ warning capability is available for prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility Fire Protection Program.

Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's Fire Protection Program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the ~~affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.~~

3/4.3.3.9 METAL IMPACT MONITORING SYSTEM

The OPERABILITY of the Metal Impact Monitoring System ensures that sufficient capability is available to detect loose metallic parts in the Reactor System

METAL IMPACT MONITORING ~~LOOSE-PART DETECTION~~ SYSTEM (Continued)

and avoid or mitigate damage to Reactor System components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE TREATMENT SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} μCi/ml are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

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SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and 10 CFR 50 Appendix G. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1). The Shearon Harris Unit 1 heatup and cooldown curves shown in Figures 3.4-~~2~~ 3 and 3.4-~~3~~ 2 AND TABLE 4.4-6 are not impacted by the 120°F limit.

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-~~2~~ 3 and 3.4-~~3~~ 2 AND TABLE 4.4-6 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-3 and 3.4-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 4 effective full power years (EFPY) of service life. The 4 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse

PRESSURE/TEMPERATURE LIMITS (Continued)

Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-~~2~~3 and 3.4-~~3~~2 include predicted adjustments for this shift in RT_{NDT} at the end of 4 EFPY as AND TABLE 4.4-6 well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75 L_a$, during performance of the periodic test, to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of -2 psig, and (2) the containment peak pressure does not exceed the design pressure of 45 psig.

The maximum peak pressure 40.9 expected using a to be obtained from a postulated main steam line break event is ~~39.1~~ psig ~~A~~ value of 1.9 psig ~~was used~~ for initial positive containment pressure. ~~in safety analyses and limits the total pressure to 41 psig, which is less than design pressure and is consistent with the safety analyses.~~ However, since the instrument tolerance for containment pressure is 1.32 psig and the high-one setpoint is 3.0 psig, the pressure limit was reduced from the high-one setpoint by slightly more than the tolerance and was set at 1.6 psig. This value will prevent spurious safety injection signals caused by instrument drift during normal operation. THE -1"wg WAS CHOSEN TO BE CONSISTENT WITH THE INITIAL ASSUMPTIONS OF THE ACCIDENT ANALYSES.

As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Fan Coolers have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. This hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," Rev. 2, November 1978.

3/4.6.5 VACUUM RELIEF ~~VALVES~~ SYSTEM

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than -1.93 psig. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of -2 psig.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

The OPERABILITY of the Emergency Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," Rev. 2, January 1976.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the Control Room Emergency Filtration System ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon a removal efficiency of 99% for elemental, particulate and organic forms of radioiodine. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

3/4.7.7 REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEM

The OPERABILITY of the Reactor Auxiliary Building Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the

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REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEM (Continued)

pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Manager-Technical Support. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has

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SEALED SOURCE CONTAMINATION (Continued)

limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources that are frequently handled are required to be tested more often than those that are not. Sealed sources that are continuously enclosed within a shielded mechanism (i.e., sealed-sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 ~~FIRE SUPPRESSION SYSTEMS~~ DELETED

~~The OPERABILITY of the Fire Suppression Systems ensures that adequate fire~~ suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the fire protection water supply and distribution system, preaction and multicycle sprinkler systems, fire hose stations, and yard fire hydrants. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met.

In the event the Fire Suppression Water System becomes inoperable, immediate ~~corrective measures must be taken since this system provides the major fire suppression capability of the plant.~~

3/4.7.11 ~~FIRE RATED ASSEMBLIES~~ DELETED

~~The functional integrity of the fire rated assemblies and barrier penetrations~~ ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are ~~a passive element in the facility Fire Protection Program and are subject to periodic inspections.~~

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~~FIRE RATED ASSEMBLIES (Continued)~~

~~Fire barrier penetrations, including cable penetration barriers, fire doors and~~ dampers ~~are~~ considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier ~~must be verified OPERABLE and an hourly fire watch patrol established until the barrier is restored to functional status~~.

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits do not include an allowance for instrument errors.

3.4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

The OPERABILITY of the Emergency Service Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

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3/4.8.1, 3/4.8.2, AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The switchyard is designed using a breaker-and-a-half scheme. The switchyard currently has ~~five~~ SIX connections with the CP&L transmission network; each of these transmission lines is physically independent. The switchyard has one connection with each of the two Startup Auxiliary Transformers and each SAT can be fed directly from an associated offsite transmission line. The Startup Auxiliary Transformers are the preferred power source for the Class 1E ESF buses. The minimum alignment of offsite power sources will be maintained such that at least two physically independent offsite circuits are available. The two physically independent circuits may consist of any two of the incoming transmission lines to the SATs (either through the switchyard or directly) and into the Class 1E system. As long as there are at least two transmission lines in service and two circuits through the SATs to the Class 1E buses, the LCO is met.

During MODES 5 and 6, the Class 1E buses can be energized from the offsite transmission net work via a combination of the main transformers, and unit auxiliary transformers. This arrangement may be used to satisfy the requirement of one physically independent circuit.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE. ~~and that the steam-driven auxiliary feedwater pump is OPERABLE.~~ This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

BASES

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements BASED UPON for demonstrating the OPERABILITY of the diesel generators are ~~in accordance with~~ the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 as modified in accordance with the guidance of IE Notice 85-32, April 22, 1985; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety

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A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers ~~and fuses~~ provide assurance of breaker ~~and fuse~~ reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker ~~and/or fuse.~~ Each manufacturer's molded case and metal case circuit breakers ~~and/or fuses~~ are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers ~~and/or fuses~~ are tested. If a wide variety exists within any manufacturer's brand of circuit breakers ~~and/or fuses~~, it is necessary to divide that manufacturer's breakers ~~and/or fuses~~ into groups and treat each group as a separate type of breaker ~~or fuses~~ for surveillance purposes.

The bypassing of the motor-operated valves thermal overload protection during accident conditions by integral bypass devices ensures that safety-related valves will not be prevented from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection during accident conditions are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

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3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND NEW AND SPENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

The limitations on the Fuel Handling Building Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	200 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential ~~< 625°F~~. GREATER THAN 350°F BUT LESS THAN 625°F.
	200 leak tests.	Pressurized to $\geq$ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 1481 psig.

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6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant General Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Foreman (or, during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President-Harris Nuclear Project shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. An individual qualified as a Radiation Control Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

~~e. A site Fire Brigade of at least five members* shall be maintained on-site at all times. The Fire Brigade shall not include any members of the minimum shift crew necessary for safe shutdown of the unit, as specified in Table 6.2-1, nor any personnel required for other essential functions during a fire emergency; and~~

*The Radiation Control Technician ~~and Fire Brigade~~ composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

CORPORATE ORGANIZATION

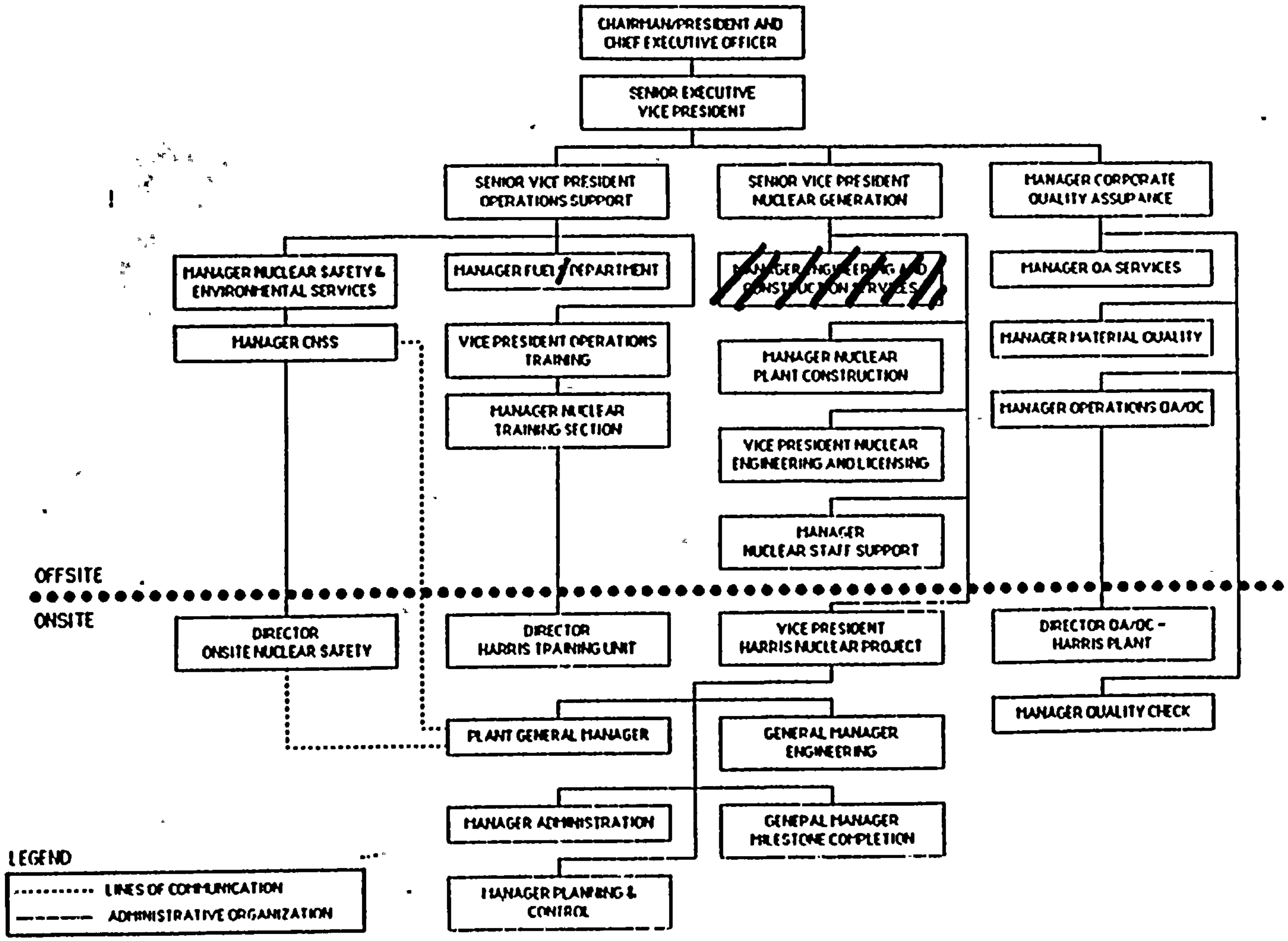

FIGURE 6.2-1

OFFSITE ORGANIZATION

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PLANT ORGANIZATION

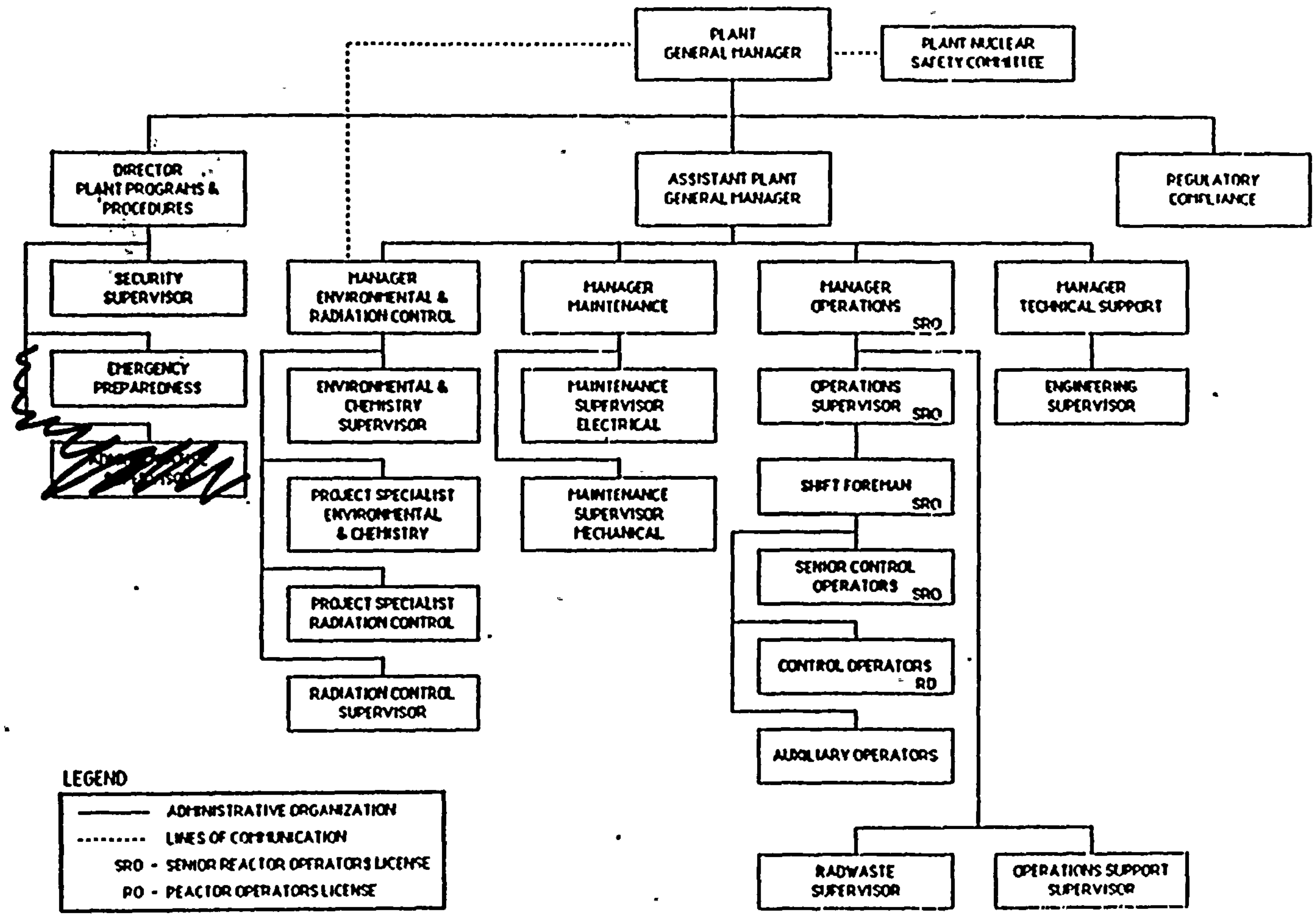

FIGURE 6.2-2

UNIT ORGANIZATION

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6.2.3 ONSITE NUCLEAR SAFETY (ONS) UNIT

FUNCTION

(INCLUDING INFORMATION FORWARDED FROM INPO FROM THEIR EVALUATION OF ALL INDUSTRY LER'S)

6.2.3.1 The ONS Unit shall function to examine unit operating characteristics, NRC issuances, industry advisories, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety, to appropriate levels of management, up to and including the Senior Vice President-Operations Support, if necessary.

COMPOSITION

6.2.3.2 The ONS Unit shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a baccalaureate degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ONS Unit shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ONS Unit shall be prepared, maintained, and forwarded each calendar month to the Manager-Nuclear Safety and Environmental Services.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Foreman in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a baccalaureate degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of the September 1979 draft of ANS 3.1, with the exceptions and alternatives noted on FSAR pages 1.8-8 (Am.20), 1.8-9 (Am.17), 1.8-10 (Am.22),

*Not responsible for sign-off function.

RESPONSIBILITIES (Continued)

b. Provide written notification within 24 hours to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services of disagreement between the PNSC and the Plant General Manager. However, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.2.8 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services.

6.5.3 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.3.1 The Corporate Nuclear Safety Section (CNSS) of the Nuclear Safety and Environmental Services Department shall function to provide independent review of plant changes, tests, and procedures; verify that REPORTABLE EVENTS are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer. They shall also evaluate all CP&L LERs ~~and other industry reports including the information forwarded by INPO from their evaluation of all industry LERs~~ for their potential applicability to other CP&L units.

ORGANIZATION

6.5.3.2 The individuals assigned responsibility for independent reviews shall be technically qualified in a specified technical discipline or disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Administrative controls,
i. Quality assurance practices,
j. Nondestructive testing, and
k. Other appropriate fields associated with the unique characteristics.

REVIEW (Continued)

e. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, ~~and~~ internal procedures or instructions having nuclear safety significance, significant operating abnormalities or deviations from normal and expected performance of SAFETY RELATED plant structures, systems, or components ~~that affect nuclear safety~~; WHICH REQUIRE WRITTEN NOTIFICATION TO THE COMMISSION.

f. All REPORTABLE EVENTS;

g. All proposed modifications that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50 or involve a change to the Technical Specifications;

h. Any other matter involving safe operation of the nuclear power plant that the Manager-Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager-Corporate Nuclear Safety Section by the onsite operating organization or other functional organizational units within Carolina Power & Light Company;

~~i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and~~

i ~~j~~. Reports and minutes of the PNSC.

6.5.3.10 Review of items considered under Specification 6.5.3.9.e, h and ~~j~~ i above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.3.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNSS files.

b. Recommendations and concerns shall be submitted to the Plant General Manager and Vice President-Harris Nuclear Project within 14 days of completion of the review. A report summarizing the reviews encompassed by Specification 6.5.3.9 shall be provided to the Plant General Manager and the Vice President-Harris Nuclear Project every other month.

c. A summation of recommendations and concerns of the Corporate Nuclear Safety Section shall be submitted to the Chairman/President and Chief Executive Officer and other appropriate senior management personnel at least every other month.

RECORDS

6.5.4.3 Records of audits shall be prepared and retained.

6.5.4.4 Audit reports encompassed by Specification 6.5.4.1 shall be prepared, approved by the Manager-Quality Assurance Services, and forwarded, within 30 days after completion of the audit, to the Executive Vice President-Power Supply and Engineering and Construction, Senior Vice President-Nuclear Generation, Vice President-Harris Nuclear Project, Manager-Nuclear Safety and Environmental Services, Plant General Manager, and the management positions responsible for the areas audited.

SENIOR

AUTHORITY

6.5.4.5 The Manager-Quality Assurance Service Section, under the Manager-Corporate Quality Assurance Department, shall be responsible for the following:

a. Administering the Corporate Quality Assurance Audit Program.

b. Approval of the individuals selected to conduct quality assurance audits.

6.5.4.6 Audit personnel shall be independent of the area audited.

6.5.4.7 Selection of personnel for auditing assignments shall be based on experience or training that establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting audit personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

6.5.4.8 Qualified outside consultants, or other individuals independent from those personnel directly involved in plant operation, shall be used to augment the audit teams when necessary.

6.5.5 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.5.1 An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months using either qualified offsite licensee personnel or an outside fire protection firm.

6.5.5.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

6.5.5.3 Copies of the audit reports and responses to them shall be forwarded to the Vice President-Harris Nuclear Project and the Manager-Corporate Quality Assurance.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

Attachment 1 to NLS-86-326

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 779 Comment Type: IMPROVEMENT

LCO Number: 3.08.04.02 Page Number: 3/4 8-39

Section Number: 4.8.4.2.a

Comment:

CHANGE "92 days" TO "18 months"

Basis

THIS CHANGE WILL PERMIT US TO PERFORM THE REQUIRED TESTING AT AN APPROPRIATE INTERVAL THAT WILL NOT REQUIRE DISRUPTION OF NORMAL PLANT OPERATIONS. WITH THE PRESENT PLANT CONFIGURATION, THIS TEST CAN ONLY BE PERFORMED BY ACTUALLY CYCLING EACH VALVE. MANY OF THE VALVES ON THIS LIST CAN NOT BE CYCLED WHILE THE PLANT IS IN OPERATION WITHOUT CREATING VERY UNDESIRABLE CONSEQUENCES IN TERMS OF PLANT CONDITIONS OR TRANSIENTS. IN ADDITION, INDIVIDUAL VALVE TESTS WILL REQUIRE MUCH MORE PERSONNEL TIME IN THE RADIATION CONTROL AREA WITH POTENTIALLY UNDESIRABLE ALARA CONSEQUENCES. THIS CHANGE HAS BEEN PREVIOUSLY DISCUSSED WITH MEMBERS OF THE NRC STAFF AND IS THE SAME FREQUENCY GRANTED TO OTHER PLANTS WITH SIMILAR BYPASS REQUIREMENTS SUCH AS MILLSTONE AND V.C. SUMMER.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 780 Comment Type: ERROR

LCO Number: 3.08.04.02 Page Number: 3/4 8-42

Section Number: TABLE 3.8-2

Comment:

DELETE VALVES 1SW-1, 1SW-2, 1SW-3 AND 1SW-4 FROM THE TABLE.

Basis

DUE TO A PLANT MODIFICATION, THESE VALVES HAVE BEEN CHANGED TO MANUAL VALVES AND THEREFORE THERE IS NO THERMAL OVERLOAD BYPASS. REMOTE OPERATION OF THESE VALVES WAS NOT ASSUMED BY ANY SAFETY ANALYSIS.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 781

Comment Type: ERROR

LCO Number: 3.05.02

Page Number: 3/4 5-6

Section Number: 4.5.2.h.1.b

Comment:

CHANGE THE VALUE "650 gpm" TO "685 gpm".

Basis

THIS VALUE IS BEING CHANGED DUE TO A REEVALUATION OF THE LIMITATIONS OF THE PUMP CURVE. ONE PUMP HAD A FLOW RATE GREATER THAN THE CURRENT 650 gpm LIMIT. IN EVALUATING THIS ITEM, IT WAS DETERMINED THAT THE CURRENT LIMIT WAS UNNECESSARILY CONSERVATIVE. THE NEW VALUE HAS BEEN DETERMINED TO BE ACCEPTABLE FOR ALL THREE PUMPS AND STILL PROVIDES ADEQUATE MARGIN TO THE ACTUAL LIMITS OF THE PUMP CURVE.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 782

Comment Type: IMPROVEMENT

LCO Number: B 3/4.01.01.01

Page Number: B 3/4 1-1

Section Number: B 3/4.1.1.1

Comment:

ADD A NEW SENTENCE AFTER THE WORDS "inadvertent dilution event." AS FOLLOWS:
The unit "pcm" is used throughout these specifications to conform with the reactivity information provided by the NSSS supplier; 1000 pcm is equal to 1% delta k/k.

Basis

THIS CHANGE IS IN RESPONSE TO AN NRC COMMENT. IT PROVIDES THE NECESSARY EQUIVALENCY INFORMATION, BUT DOES NOT CONFUSE THE ACTUAL SPECIFICATION.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 783 Comment Type: IMPROVEMENT

LCO Number: 3.03.01 Page Number: 3/4 3-2 & 4

Section Number: TABLE 3.3-1

Comment:

INSERT THE PHRASE "(above P-7)" AFTER THE FUNCTIONAL UNIT DESCRIPTION FOR ITEMS 9 & 11 ON PAGE 3/4 3-2 AND ITEM 17 ON PAGE 3/4 3-4.

Basis

THIS CHANGE IS FOR CONSISTENCY AND CLARITY. THE DEPENDENCE IS ALREADY ACCURATELY DESCRIBED IN THE BASES FOR SECTION 2. OTHER ITEMS IN THIS TABLE HAVE SIMILAR NOTATIONS BUT THESE NOTATIONS HAVE BEEN PREVIOUSLY OVERLOOKED.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 784 Comment Type: ERROR

LCO Number: 3.03.01 Page Number: 3/4 3-3

Section Number: TABLE 3.3-1

Comment:

IN ITEM 13 CHANGE THE MINIMUM CHANNELS OPERABLE FROM "3 per" TO "2 per".

Basis

THE CORRECT VALUE ON THIS ITEM FOR SHNPP IS 2. THE CURRENT VALUE HAS BEEN ERRONEOUSLY LEFT FROM THE ORIGINAL STS GENERIC VALUE. SHNPP STILL USES THE OLDER INSTRUMENTATION LOGIC ON STEAM GENERATOR LEVEL. THIS LOGIC IS THE SAME AS THAT USED FOR V.C. SUMMER AND MOST OLDER WESTINGHOUSE PLANTS IN WHICH 2 HAS ALWAYS BEEN THE MINIMUM REQUIREMENT FOR THAT LOGIC.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 785 | Comment Type: IMPROVEMENT

LCO Number: 3.03.01 | Page Number: 3/4 3-7

Section Number: TABLE 3.3-1

Comment:

IN THE NOTE 5 - MAKE THE EXISTING NOTE PART "a", AND ADD A NEW PART "b" NOTE AS FOLLOWS:

b. With no channels OPERABLE, open the Reactor Trip Breakers within 1 hour and suspend all operations involving positive reactivity changes. Verifiy compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

Basis

THIS CHANGE PROVIDES FOR A MORE COMPLETE ACTION STATEMENT WHICH COVERS ALL POSSIBILITIES. THIS IS DESIRABLE SINCE THE ACTIONS OF 3.0.3 MAY NOT BE MEANINGFUL IN THIS CASE.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 786

Comment Type: ERROR

LCO Number: 3.03.02

Page Number: 3/4 3-23

Section Number: TABLE 3.3-3

Comment:

IN ITEM 6f - CHANGE THE ACTION FROM "18" TO "15*".

Basis

THE CURRENT NOTE IS INAPPROPRIATE FOR THE SHNPP DESIGN LOGIC. THE MINIMUM AND TOTAL ARE THE SAME BUT WE HAVE SHOWN THE LOGIC ON A PER PUMP BASIS. BY GOING TO ACTION 15*, WE PRESERVE THE ORIGINAL LEVEL OF PROTECTION BY TRIPPING THE FAILED CHANNEL WHICH ALSO JUSTIFIES CONTINUED OPERATION.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 787 Comment Type: ERROR

LCO Number: 3.03.01 Page Number: 3/4 3-2 & 3

Section Number: TABLE 3.3-1

Comment:

ON PAGE 3/4 3-2 ITEM 9 - ADD NOTE "(1)" TO THE ACTION COLUMN.
ON PAGE 3/4 3-3 ITEM 15 - DELETE NOTE "(1)" FROM THE ACTION COLUMN.

Basis

A CONSISTENCY CHECK HAS SHOWN THAT WHILE ITEM 9 DOES HAVE MORE RESTRICTIVE REQUIREMENTS IN TABLE 3.3-3, ITEM 15 IS NOT LISTED IN THAT TABLE AT ALL. THEREFORE THIS CHANGE IS NECESSARY.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 788 | Comment Type: IMPROVEMENT

LCO Number: 3.03.02 | Page Number: 3/4 3-31 & 36

Section Number: TABLE 3.3-4

Comment:

ON PAGE 3/4 3-31 ITEM 4e - DELETE THE "/s" FROM THE TRIP SETPOINT AND ALLOWABLE VALUE COLUMNS AND REPLACE IT WITH A "#".

ON PAGE 3/4 3-36 ADD A NEW FOOTNOTE # AS FOLLOWS:
The indicated values are the effective, cumulative, rate compensated pressure drops as seen by the comparator.

Basis

THIS CHANGE IS NEEDED TO PREVENT POSSIBLE MISCONCEPTION ABOUT THESE SETPOINTS. THIS ITEM IS A RATE COMPENSATED VARIABLE FUNCTION. IT WILL TRIP AT A PRESSURE DECREASE OF 100psi/s FOR 1 SECOND, BUT IT WILL ALSO BE TRIPPED BY SLOWER RATES OVER LONGER PERIODS OF TIME (DOWN TO APPROXIMATELY 2psi/s FOR 4 MINUTES). THE PROPOSED WORDING WILL INDICATE THAT THIS IS NOT A SIMPLE SETPOINT AND WILL ALERT THOSE WHO READ IT THAT THEY MUST GET MORE DETAIL TO FULLY UNDERSTAND THIS RATE FUNCTION.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 789

Comment Type: IMPROVEMENT

LCO Number: 3.02.03

Page Number: 3/4 2-10

Section Number: 4.2.3.5

Comment:

CHANGE "7 days" TO "30 days".

Basis

THE CURRENT 7 DAY INTERVAL WILL BE EXTREMELY DIFFICULT TO MEET. INVESTIGATION OF THE ISSUE DISCLOSED THAT 7 DAYS WAS ORIGINALLY PROPOSED BY UTILITIES AS A VERY CONSERVATIVE TIME TO ENSURE THAT NO INSTRUMENT DRIFT COULD AFFECT THE TEST RESULTS. SEVERAL YEARS OF TEST RESULTS HAVE SHOWN THIS ORIGINAL CONCERN TO BE UNFOUNDED IF THE INSTRUMENTS ARE SIMPLY WITHIN THE MANUFACTURER'S RECOMMENDED CALIBRATION FREQUENCY. WITHIN THIS FREQUENCY, THE NOMINAL INSTRUMENT ACCURACY, WHICH IS FULLY ACCOUNTED FOR IN THE ANALYSIS, ALREADY INCLUDES AN ALLOWANCE FOR WHATEVER MINOR DRIFT MAY OCCUR. THE 30 DAY REQUEST IS STILL ONLY A FRACTION OF EVEN THE SHORTEST RECOMMENDED CALIBRATION FREQUENCY. THIS WILL PERMIT A MUCH MORE REASONABLE MANPOWER ALLOCATION FOR THE CALIBRATIONS WITHOUT ANY ADVERSE AFFECT ON MEASUREMENT ACCURACY.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 790 Comment Type: ERROR

LCO Number: 3.06.02.03 Page Number: 3/4 6-13

Section Number: 4.6.2.3.a.1

Comment:

CHANGE THE VALUE "1500 gpm" TO "1425 gpm".

Basis

PLANT PREOPERATIONAL TEST DATA EVALUATION HAS LED TO A REEXAMINATION OF THIS REQUIREMENT. IT HAS BEEN DETERMINED THAT 1425 gpm IS THE APPROPRIATE VALUE FOR THIS SPECIFICATION. THIS VALUE IS CONSISTENT WITH THE SAFETY ANALYSIS FOR THE PLANT.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 791 Comment Type: ERROR

LCO Number: B 3/4.06.02.02 Page Number: B 3/4 6-3

Section Number: B 3/4.6.2.2

Comment:

DELETE THE LAST SENTENCE OF THE BASES PARAGRAPH 3/4.6.2.2 ON THE SPRAY ADDITIVE SYSTEM AND REPLACE IT WITH:
"The RWST level of 436,000 gallons provides adequate test conditions to demonstrate that the flow rate is within the maximum and minimum assumptions of the analyses."

Basis

THIS CHANGE IS NECESSARY TO BE CONSISTENT WITH THE CURRENT WORDING OF THE SPECIFICATION. THE SPECIFICATION WAS CHANGED IN JULY AND THE CHANGE HAS BEEN AGREED TO BY THE NRR STAFF.

CP&L Comments

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Record Number: 792

Comment Type: ERROR

LCO Number: 3.07.01.05

Page Number: 3/4 7-9

Section Number: 4.7.1.5

Comment:

IN 4.7.1.5 CHANGE "MODE 3" TO "MODES 3 or 4."

Basis

A CHANGE TO SHOW MODE 4 FOR THE MSIV's WAS MADE SOME TIME AGO TO RESOLVE A LONG STANDING CONFLICT WITH THE STS. HOWEVER, IT IS STILL NOT POSSIBLE TO PROPERLY TEST THE VALVES UNTIL THERE IS SUFFICENT STEAM PRESSURE. THE STS HAS ALWAYS GRANTED THE 4.0.4 EXEMPTION FOR MODE 3 AND THIS IS SIMPLY A LOGICAL ENTENSION TO THE LOWER MODE.

CP&L Comments

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Record Number: 793 | Comment Type: IMPROVEMENT

LCO Number: 6.03 & INDEX | Page Number: 6-6 & xviii

Section Number: 6.3 & INDEX

Comment:

DELETE THE SECTION 6.3.1 AND MARK THE SECTION AS "DELETED".

ALSO ON PAGE xviii CHANGE "UNIT STAFF QUALIFICATIONS" TO "DELETED".

Basis

THE INFORMATION IN THIS PARAGRAPH IS COVERED IN MUCH MORE DETAIL IN THE FSAR. AS CURRENTLY WRITTEN, THIS MAY REQUIRE FREQUENT TECH SPEC CHANGES FOR PURELY ADMINISTRATIVE REASONS. THIS SPECIFICATION IS BEING DELETED IN THE FORTHCOMING SEABROOK TECHNICAL SPECIFICATIONS. THIS CHANGE HAS BEEN PREVIOUSLY DISCUSSED WITH NRR STAFF.

CP&L Comments

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Record Number: 794 Comment Type: ERROR

LCO Number: 6.02.01 Page Number: 6-3

Section Number: FIGURE 6.2-1

Comment:

CHANGE THE FIGURE PER THE ATTACHED.

Basis

THIS MARKUP REFLECTS RECENTLY ANNOUNCED CHANGES IN THE CORPORATE STRUCTURE OF CP&L.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 795 Comment Type: ERROR

LCO Number: 3.07.07 & 3.09.12 Page Number: 3/4 7-17 & 9-14

Section Number: 4.7.7 & 4.9.12

Comment:

IN ITEMS 4.7.7.b.1 (P 7-17) and 4.9.12.b.1 (P 9-14) - CHANGE "0.05%" TO "0.05% HEPA 1.0% CHARCOAL
4.7.7.f (P 7-18) and 4.9.12.f (P 9-16) - CHANGE "0.05%" TO "1.0 %".

Basis

THE FILTERS COVERED BY THESE TWO SPECIFICATIONS ARE 95% EFFICIENT. ACCORDING TO GENERIC LETTER 83-13, MARCH 2, 1983, A VALUE OF 1.0% IS APPROPRIATE FOR FILTERS ASSUMED TO BE 95% EFFICIENT. THE INCORRECT VALUE WAS ERRONEOUSLY SUBMITTED BY CP&L.

CP&L Comments

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Record Number: 796 Comment Type: ERROR

LCO Number: 3.06.03 Page Number: 3/4 6-19 & 19a

Section Number: TABLE 3.6-1

Comment:

ON PAGE 3/4 6-19 FOR PENETRATION 73A AND 73B - DELETE THE WORDS "RAD MONITOR &" FROM THE FUNCTION AND ADD "3" TO THE APPLICABLE NOTES COLUMN. ADDITIONALLY, INSERT THE ENTRIES FOR PENETRATIONS 83A AND 83B FROM PAGE 3/4 6-19a IN THE INDICATED PLACE ON PAGE 3/4 6-19,

Basis

A PLANT MODIFICATION HAS SEPARATED THE RADIATION MONITOR AND HYDROGEN ANALYZER LINES TO PROVIDE FOR MORE EFFIEIENT OPERATION. THIS CHANGE REFLECTS THE DESIGN CHANGE. THE ADDITION OF NOTE 3 IS NEEDED TO ACKNOWLEDGE THAT THE HYDROGEN MONITORS MAY BE OPENED UNDER ADMINISTRATIVE CONTROL TO PERMIT SAMPLING IN A POST-ACCIDENT SITUATION.

CP&L Comments

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Record Number: 797

Comment Type: ERROR

LCO Number: 3.06.03

Page Number: 3/4 6-20 - 28

Section Number: TABLE 3.6-1

Comment:

PAGE 3/4 6-20 CHANGE
CP-B1 TO CP-9
CP-B3 TO CP-10
CP-B4 TO CP-7
CP-B2 TO CP-6
CP-B7 TO CP-4

PAGE 3/4 6-21 CHANGE
CP-B5 TO CP-5
CP-B8 TO CP-1
CP-B6 TO CP-3
CB-B1 TO CB-2
CB-B2 TO CB-6

PAGE 3/4 6-25 CHANGE
CM-B5 TO CM-2

PAGE 3/4 6-26 CHANGE
CB-V1 TO CB-3
CM-V1 TO CM-7

PAGE 3/4 6-27 CHANGE
CM-B6 TO CM-5
CM-B4 TO CM-4

PAGE 3/4 6-28 CHANGE
CB-V2 TO CB-7

Basis

THESE NUMBERS REPRESENT CP&L DESIGNATORS FOR THE VALVES. DUE TO A RECENT ADMINISTRATIVE CHANGE, THE VALVE NUMBERS HAVE BEEN REVISED.

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 798 Comment Type: IMPROVEMENT

LCO Number: 3.05.02 Page Number: 3/4 5-5

Section Number: 4.5.2.f.2

Comment:

CHANGE ITEM f.2 TO THE FOLLOWING:
RHR pump greater than or equal to 100 psid at a flow rate of at least 3663 gpm.

Basis

THIS CHANGE IS NEEDED TO REDUCE REDUNDANT TESTING REQUIREMENTS. THE CURRENT dp VALUE IS APPROPRIATE ONLY FOR OPERATION NEAR SHUTOFF HEAD. OTHER TESTING CRITERIA IMPOSED BY THE STAFF FOR CHECK VALVES REQUIRE THAT WE TEST THE PUMPS AT OR NEAR FULL FLOW. BY CHANGING THIS SPECIFICATION TO TEST AT A DIFFERENT POINT ON THE PUMP CURVE, THIS REDUNDANCY CAN BE AVOIDED. TESTING AT THIS POINT IS NOT ONLY STILL CONSISTENT WITH THE SAFETY ANALYSIS, IT IS ACTUALLY A BETTER TEST BECAUSE IT MORE CLOSELY APPROACHES THE ASSUMPTIONS MADE IN THE ANALYSIS.

Attachment 2 to NLS-86-326

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LIMITING CONDITION FOR OPERATION

ACTION (Continued):

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within acceptable limits prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,
   2. A nominal 75% of RATED THERMAL POWER, and
   3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within acceptable limits:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the acceptable limit:

a. At least once per 12 hours by the use of main control board instrumentation or equivalent, and
b. At least once per 31 days by the use of process computer readings or digital voltmeter measurement.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. The measurement instrumentation shall be calibrated within ~~7~~ 30 days prior to the performance of the calorimetric flow measurement.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature ΔT	3	2	2	1, 2	6#
8. Overpower ΔT	3	2	2	1, 2	6#
9. Pressurizer Pressure--Low (Above P-7)	3	2	2	1	6# (1)
10. Pressurizer Pressure--High	3	2	2	1, 2	6#
11. Pressurizer Water Level--High (Above P-7)	3	2	2	1	6#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2 ~~3~~/stm. gen. each operating stm. gen.	1, 2	6#(1)
14. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feedwater flow mismatch in each stm. gen.	1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feedwater flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6#
15. Undervoltage--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Underfrequency--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6#
17. Turbine Trip (Above P-7)					
a. Low Fluid Oil Pressure	3	2	2	1	6#
b. Turbine Throttle Valve Closure	4	4	1	1	10#
18. Safety Injection Input from ESF	2	1	2	1, 2	8
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1	7
or					
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	7

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, and verify compliance with the shutdown margin requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

b. With no channels OPERABLE, open the Reactor Trip Breakers within ~~one~~ 1 hour and suspend all operations involving positive Reactivity changes. ~~and~~ Verify compliance with the SHUTDOWN MARGIN ~~within~~ Requirements of Specification 3.1.1.1 or 3.1.1.2, as Applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	~~22~~ 23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Steam Generator Water Level--Low-Low					
1) Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. in any stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
2) Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. in any 2 stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for Loss of Offsite Power initiating functions and requirements.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	1/pump	1/pump	1/pump	1, 2	~~18~~ 15*

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Main Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	2.7	0.71	1.5	≤3.0 psig	≤3.6 psig
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Trip Setpoints and Allowable Values.				
e. Negative Steam Line Pressure Rate--High	2.3	0.5	0	≤ 100 psi#	≤ 122.8 psi#**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \geq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate--High is less than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

The indicated values are the effective, cumulative, rate compensated pressure drops as seen by the comparator.

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SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on safety injection actuation test signal and on safety injection switchover to containment sump from an RWST Lo-Lo level test signal, and

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a) charging/safety injection pump,
b) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure when tested pursuant to Specification 4.0.5:

1. charging/safety injection pump (Refer to Specification 4.1.2.4)
2. RHR pump ≥ ~~134~~ 100 psid at a flow rate of at least 3663 gpm.

g. By verifying that the locking mechanism is in place and locked for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2. At least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

~~HPSI SYSTEM EBASCO Valve No. (FSAR)~~	HPSI SYSTEM EBASCO Valve No.	HPSI SYSTEM CP&L Valve No.
~~2SI-V16SA-1~~	2SI-V440SA-1	1SI-5
~~2SI-V22SB-1~~	2SI-V439SB-1	1SI-6
~~2SI-V28SA-1~~	2SI-V438SA-1	1SI-7
~~2SI-V62SA-1~~	2SI-V437SA-1	1SI-69
~~2SI-V68SB-1~~	2SI-V436SB-1	1SI-70
~~2SI-V74SA-1~~	2SI-V435SA-1	1SI-71
~~2SI-V38SA-1~~	2SI-V434SA-1	1SI-101
~~2SI-V44SB-1~~	2SI-V433SB-1	1SI-102
~~2SI-V50SA-1~~	2SI-V432SA-1	1SI-103
~~2SI-V83SA-1~~	2SI-V431SA-1	1SI-124
~~2SI-V89SB-1~~	2SI-V430SB-1	1SI-125
~~2SI-V95SA-1~~	2SI-V429SA-1	1SI-126

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For charging/safety injection pump lines, with a single pump running:

   a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 379 gpm, and

   b) The total pump flow rate is less than or equal to ~~650~~ 685 gpm.

2. For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

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CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Four containment fan coolers (AH-1, AH-2, AH-3 and AH-4) shall be OPERABLE with one of two fans in each cooler capable of operation at low speed. Train SA consists of AH-2 and AH-3. Train SB consists of AH-1 and AH-4.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required trains of fan coolers to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one train of the above required containment fan coolers inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable train of containment fan coolers to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment fan coolers shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each fan train from the control room, and verifying that each fan train operates for at least 15 minutes, and

2. Verifying a cooling water flow rate of greater than or equal to ~~1500~~ gpm to each cooler.
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b. At least once per 18 months by verifying that each fan train starts automatically on a safety injection test signal.

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
109	1AF-159 (AF-V164)	AUX. F.W. TO S/G B (HYDRAZINE)	10	1,2,6
109	1AF-157 (AF-V165)	AUX. F.W. TO S/G B (AMMONIA)	10	1,2,6
110	1AF-163 (AF-V166)	AUX. F.W. TO S/G C (HYDRAZINE)	10	1,2,6
110	1AF-161 (AF-V167)	AUX. F.W. TO S/G C (AMMONIA)	10	1,2,6
73A	1SP-12 (SP-V300)	~~RAD MONITOR &~~ H_2 ANALYZER	<60	2,3
73A	1SP-915 (SP-V348)	~~RAD MONITOR &~~ H_2 ANALYZER	<60	2,3
73B	1SP-941 (SP-V301)	~~RAD MONITOR &~~ H_2 ANALYZER	<60	2,3
73B	1SP-917 (SP-V349)	~~RAD MONITOR &~~ H_2 ANALYZER	<60	2,3
86A	1SP-42 (SP-V308)	HYDROGEN ANALYZER	<60	2,3
86A	1SP-919 (SP-V314)	HYDROGEN ANALYZER	<60	2,3
86B	1SP-62 (SP-V309)	HYDROGEN ANALYZER	<60	2,3
86B	1SP-943 (SP-V315)	HYDROGEN ANALYZER	<60	2,3

Insert 4 Valves from P 3/4 6-19a

2. PHASE B ISOLATION

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
35	1CC-208 (CC-V170)	CCW TO RCP	10	N/A
36	1CC-297 (CC-V184)	CCW FROM RCP	10	N/A
36	1CC-299 (CC-V183)	CCW FROM RCP	10	N/A

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAX ISOL TIME(SEC)	APP. NOTES
83 A	1SP-916 (SP-V448)	RADIATION MONITOR	<60	2
83A	1SP-16 (SP-V449)	RADIATION MONITOR	<60	2
83B	1SP-918 (SP-V450)	RADIATION MONITOR	<60	2
83 B	1SP-939 (SP-V451)	RADIATION MONITOR	≤60	2

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
39	1CC-249 (CC-V191)	CCW FROM RCP THERMAL BARRIERS	10	N/A
39	1CC-251 (CC-V190)	CCW FROM RCP THERMAL BARRIERS	10	N/A
3. SAFETY INJECTION ACTUATION				
8	1CS-238 (CS-V610)	CVCS NORMAL CHARGING	10	N/A
51	1BD-11 (BD-V11)	S/G A BLOWDOWN	<60	1,2,6
52	1BD-30 (BD-V15)	S/G B BLOWDOWN	<60	1,2,6
53	1BD-49 (BD-V19)	S/G C BLOWDOWN	<60	1,2,6
54	1SP-217 (SP-V120)	S/G A SAMPLE	<60	1,2,6
55	1SP-222 (SP-V121)	S/G B SAMPLE	<60	1,2,6
56	1SP-227 (SP-V122)	S/G C SAMPLE	<60	1,2,6
4. CONTAINMENT VENTILATION ISOLATION				
57	CP-9 (CP-B1)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5
57	CP-10 (CP-B3)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	2,5
57	CP-7 (CP-B4)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	2,5
57	CP-6 (CP-B2)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5
58	CP-4 (CP-B7)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	2,5

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
58	CP-5 (CP-B5)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5
58	CP-1 (CP-B8)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	2,5
58	CP-3 (CP-B6)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5
59	CB-2 (CB-B1)	CONTAINMENT VACUUM RELIEF	5	3
98	CB-6 (CB-B2)	CONTAINMENT VACUUM RELIEF	5	3
5. CONTAINMENT SPRAY ACTUATION				
23	1CT-50 (CT-V21)	CONTAINMENT SPRAY	N/A	3
24	1CT-88 (CT-V43)	CONTAINMENT SPRAY	N/A	3
6. MAIN STEAM LINE ISOLATION				
3	1MS-80 (MS-V1)	MSIV (S/G A)	5	1,4
3	1MS-81 (MS-F1)	MSIV BYPASS	10	1,2,3,6
3	1MS-231 (MS-V59)	MS DRAIN TO CONDENSER	<60	1,2,6
2	1MS-82 (MS-V2)	MSIV (S/G B)	5	1,4
2	1MS-83 (MS-F2)	MSIV BYPASS	10	1,2,3,6
2	1MS-266 (MS-V60)	MS DRAIN TO CONDENSER	<60	1,2,6
1	1MS-84 (MS-V3)	MSIV (S/G C)	5	1,~~3~~4

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
32	1SW-110 (SW-B50)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1,6
17	1SI-3 (SI-V505)	SI TO HIGH HEAD COLD LEG	N/A	3
17	1SI-4 (SI-V506)	SI TO HIGH HEAD COLD LEG	N/A	3
2	1MS-70 (MS-V8)	MAIN STEAM B TO AUXILIARY F.W. TURBINE	N/A	1,3,6
1	1MS-72 (MS-V9)	MAIN STEAM C TO AUXILIARY F.W. TURBINE	N/A	1,3,6
63	CM-2 (CM-B5)	H_2 PURGE EXHAUST	N/A	3

10. MANUAL VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
17	1SI-43 (SI-V30)	SI-HIGH HEAD TO COLD LEGS	N/A	1,3
34	1LT-6 (LT-V2)	ILRT ROTOMETER (LOCKED CLOSED)	N/A	2,3
41	1SA-80 (SA-V14)	SERVICE AIR (LOCKED CLOSED)	N/A	2,3
42	1ED-119 (WL-D651)	RCDT PUMP DISCH BYPASS (LOCKED CLOSED)	N/A	2,3
44	1SF-145 (SF-D164)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
44	1SF-144 (SF-D165)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
45	1SF-118 (SF-D25)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
45	1SF-119 (SF-D26)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
39	1CC-250 (CC-V50)	CCW FROM RCP THERMAL BARRIER	N/A	N/A

MOVE THIS VALVE UNDER ITEM
FOR CHECK VALVES.

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

ALL VALVES ON THIS PAGE SHOULD BE MOVED UNDER ITEM 11 FOR CHECK VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
40	1RC-164 (RC-V525)	DEMIN WATER TO PRT	N/A	N/A
41	1SA-82 (SA-V15)	SERVICE AIR	N/A	N/A
59	CB-3 (CB-V1)	CONTAINMENT VACUUM RELIEF	N/A	N/A
61	CM-7 (CM-V1)	H_2 PURGE MAKEUP	N/A	N/A
76A	1SI-182 (SI-V150)	ACCUMULATORY FILL FROM RWST	N/A	N/A
77A	1SI-290 (SI-V188)	N_2 TO ACCUMULATORS	N/A	N/A
79	1FP-357 (FP-V48)	FIRE WATER STANDPIPE SUPPLY	N/A	N/A
80	1AI-220 (AI-V33)	INSTRUMENT AIR SUPPLY	N/A	N/A
90	1DW-65 (DW-V121)	DEMIN WATER SUPPLY	N/A	N/A
92	1SW-233 (SW-V142)	SERVICE WATER TO NNS FAN COILS	N/A	N/A
94A	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
94B	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
94C	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
95A	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
95B	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
61	CM-~~B6~~ 5 (CM-B6)	H_2 PURGE MAKEUP (LOCKED CLOSED)	N/A	3
79	1FP-355 (FP-V44)	FIRE WATER STANDPIPE SUPPLY	N/A	2,3
62	1LT-10 (LT-V4)	ILRT (LOCKED CLOSED)	N/A	2,3
63	CM-~~B7~~ 4 (CM-B4)	H_2 PURGE EXHAUST (LOCKED CLOSED)	N/A	3
90	1DW-63 (DW-V120)	DEMIN WATER SUPPLY (LOCKED CLOSED)	N/A	2,3
96	1LT-4 (LT-V1)	ILRT (LOCKED CLOSED)	N/A	2,3
108	1AF-174 (AF-V189)	WET LAY-UP TO STM GEN A AF HEADER	N/A	1,2,6
109	1AF-173 (AF-V190)	WET LAY-UP TO STM GEN B AF HEADER	N/A	1,2,6
110	1AF-175 (AF-V191)	WET LAY-UP TO STM GEN C AF HEADER	N/A	1,2,6

11. CHECK VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
8	1CS-477 (CS-V515)	CVCS NORMAL CHARGING	N/A	N/A
12	1CS-471 (CS-V67)	CVCS SEAL WATER RETURN & EXCESS LETDOWN	N/A	N/A
23	1CT-53 (CT-V27)	CONTAINMENT SPRAY TRAIN A	N/A	N/A
24	1CT-91 (CT-V51)	CONTAINMENT SPRAY TRAIN B	N/A	N/A
35	1CC-211 (CC-V171)	CCW TO RCP	N/A	N/A
36	1CC-298 (CC-V51)	CCW FROM RCP	N/A	N/A

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
98	CB-~~V2~~ 7 (CB-V2)	CONTAINMENT VACUUM RELIEF	N/A	N/A
105	1FP-349 (FP-V46)	FIRE WATER SPRINKLER SUPPLY	N/A	N/A
12. RELIEF VALVES				
7	1CS-10 (CS-R500)	CVCS NORMAL LETDOWN	N/A	N/A
15	1RH-7 (RH-R501)	RHR SUCTION FROM HOT LEG	N/A	1
16	1RH-45 (RH-R500)	RHR SUCTION FROM HOT LEG	N/A	1
29	1SW-95 (SW-R1)	SERVICE WATER FROM FAN COOLER AH-3	N/A	1
30	1SW-107 (SW-R3)	SERVICE WATER FROM FAN COOLER AH-2	N/A	1
31	1SW-96 (SW-R2)	SERVICE WATER FROM FAN COOLER AH-1	N/A	1
32	1SW-108 (SW-R4)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1

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PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODEs 3 or 4.

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3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent RAB Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one RAB Emergency Exhaust System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each RAB Emergency Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   HEPA, 1% charcoal

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6800 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980;

   2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,

meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is less than 4.1 inches water gauge while operating the unit at a flow rate of 6800 cfm ± 10%,

2. Verifying that the system starts on a safety injection test signal,

3. Verifying that the system maintains the areas served by the exhaust system at a negative pressure of greater than or equal to 1/8 inch water gauge relative to the outside atmosphere,

4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-~~1975~~ 1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the unit at a flow rate of 6800 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than ~~0.05%~~ 1.0% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6800 cfm ± 10%.

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ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve given in Table 3.8-2 shall be bypassed only under accident conditions by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not capable of being bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The thermal overload protection for the above required valves shall be verified to be bypassed only under accident conditions by an OPERABLE integral bypass device by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry:

a. At least once per ~~92 days~~ 18 months for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions; and

b. Following maintenance on the motor starter.

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	~~BYPASS DEVICE (YES/NO)~~
1MS-72 (2MS-V9)	AFWTD STEAM C ISOLATION *	~~YES~~
1SW-39 (3SW-B5)	NORMAL SW HDR A ISOLATION	~~YES~~
1SW-276 (3SW-B8)	NORMAL SW HDR A RETURN ISOL	~~YES~~
1SW-270 (3SW-B15)	SW HDR A TO AUX RSVR ISOL	~~YES~~
1SW-40 (3SW-B6)	NORMAL SW HDR B ISOL	~~YES~~
1SW-275 (3SW-B13)	SW HDR A RETURN ISOL	~~YES~~
1SW-274 (3SW-B14)	SW HDR B RETURN ISOL	~~YES~~
1SW-271 (3SW-B16)	SW HDR B TO AUX RSVR ISOL	~~YES~~
~~1SW-3 (3SW-B3)~~	~~EMER SW PUMP 1A MAIN RSVR INLET~~	~~YES~~
~~1SW-4 (3SW-B4)~~	~~EMER SW PUMP 1B MAIN RSVR INLET~~	~~YES~~
~~1SW-1 (3SW-B1)~~	~~EMER SW PUMP 1A AUX RSVR INLET~~	~~YES~~
~~1SW-2 (3SW-B2)~~	~~EMER SW PUMP 1B AUX RSVR INLET~~	~~YES~~
1SW-92 (2SW-B46)	SW TO FAN CLR AH3 INLET	~~YES~~
1SW-97 (2SW-B47)	SW [redacted] FAN CLR AH3 OUTLET	~~YES~~
1SW-91 (2SW-B45)	SW TO FAN CLR AH2 INLET	~~YES~~
1SW-109 (2SW-B49)	SW [redacted] FAN CLR AH2 OUTLET	~~YES~~
1SW-225 (2SW-B52) FROM	SW TO FAN CLR AH1 INLET	~~YES~~
1SW-98 (2SW-B48)	SW [redacted] FAN CLR AH1 OUTLET	~~YES~~
1SW-227 (2SW-B51)	SW TO FAN CLR AH4 INLET	~~YES~~
1SW-110 (2SW-B50)	SW [redacted] FAN CLR AH4 OUTLET	~~YES~~
1SW-124 (3SW-B70)	SW TO AFWTD PUMP	~~YES~~
1SW-126 (3SW-B71)	SW TO AFWTD PUMP	~~YES~~
1SW-129 (3SW-B73)	SW TO AFWTD PUMP	~~YES~~
1SW-127 (3SW-B72)	SW TO AFWTD PUMP	~~YES~~
1SW-123 (3SW-B75)	SW TO AFW PUMP A SUPPLY	~~YES~~
1SW-121 (3SW-B74)	SW TO AFW PUMP A SUPPLY	~~YES~~
1SW-132 (3SW-B77)	SW TO AFW PUMP B SUPPLY	~~YES~~
1SW-130 (3SW-B76)	SW TO AFW PUMP B SUPPLY	~~YES~~
1ED-94 (2MD-V36)	CNMT SUMP ISOLATION	~~YES~~
1ED-95 (2MD-V77)	CNMT SUMP ISOLATION	~~YES~~
3CZ-B5	RAB ELEC PROT INLET	~~YES~~
3CZ-B6	RAB ELEC PROT INLET	~~YES~~
3CZ-B7	RAB ELEC PROT EXHAUST	~~YES~~
3CZ-B8	RAB ELEC PROT EXHAUST	~~YES~~
3CZ-B32	RAB ELEC PROT PURGE MAKE-UP	~~YES~~
3CZ-B33	RAB ELEC PROT PURGE MAKE-UP	~~YES~~
3CZ-B34	RAB ELEC PROT PURGE INLET	~~YES~~
3CZ-B35	RAB ELEC PROT PURGE INLET	~~YES~~
3FV-B2	FUEL HANDLING EXHAUST INLET *	~~NO*~~
3FV-B4	FUEL HANDLING EXHAUST INLET *	~~NO*~~
3CZ-B1	CONTROL ROOM NORMAL SUPPLY ISOL *	~~NO*~~
3CZ-B3	CONTROL ROOM NORMAL EXHAUST ISOL *	~~NO*~~
3CZ-B17	CONTROL ROOM PURGE MAKE UP *	~~NO*~~
3CZ-B2	CONTROL ROOM NORMAL SUPPLY ISOL *	~~NO*~~
3CZ-B4	CONTROL ROOM EXHAUST ISOLATION *	~~NO*~~
3CZ-B18	CONTROL ROOM PURGE MAKE UP *	~~NO*~~
3CZ-B14	CONTROL ROOM PURGE EXHAUST *	~~NO*~~

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3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Emergency Exhaust System Trains shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in a storage pool.

ACTION:

a. With one Fuel Handling Building Emergency Exhaust System Train inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Handling Building Emergency Exhaust System Train is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorber.

b. With no Fuel Handling Building Emergency Exhaust System Trains OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Handling Building Emergency Exhaust System Train is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Emergency Exhaust System trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   HEPA, 1% charcoal

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6600 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.12 (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than ~~0.05%~~ in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6600 cfm ± 10%. 1.0%

REFUELING OPERATIONS

FUEL HANDLING BUILDING EMERGENCY EXHAUST

SURVEILLANCE REQUIREMENTS (Continued)

4.9.12 (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than ~~0.05%~~ 1.0% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6600 cfm ± 10%.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1770 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal, but a 2000 pcm SHUTDOWN MARGIN is required to provide adequate protection for postulated inadvertent dilution events.

Analysis of inadvertent boron dilution at cold shutdown is based on:

1. all RCCA's in the core while the RCS, except the reactor vessel, is drained (i.e., not filled), and
2. all RCCA's, except shutdown banks C and D, are fully inserted in the core while the RCS is filled.

In addition, by assuming the most reactive control rod is stuck out of the core, its worth is effectively added to the 2000 pcm shutdown margin in calculating the necessary soluble boron concentration.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; i.e., the positive limit is based on core conditions for all rods withdrawn, BOL, hot zero THERMAL POWER, and the negative limit is based on core conditions for all rods withdrawn, EOL, RATED THERMAL POWER. Accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The unit "pcm" is used throughout these specifications to conform with the reactivity information provided by the NSSS supplier; 1000 pcm is equal to 1% $\Delta k/k$.

CONTAINMENT VENTILATION SYSTEM (Continued)

gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Fan Coolers are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses. ~~With 100,000 gallons of water in the RWST, sufficient head pressure, approximately 70 feet of water, is available at the eductor.~~ The RWST level of 436,000 gallons provides adequate test conditions to demonstrate that the flow rate is within the maximum and minimum ~~flow~~ assumptions of the analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Fan Coolers ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Fan Coolers and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere.

CORPORATE ORGANIZATION

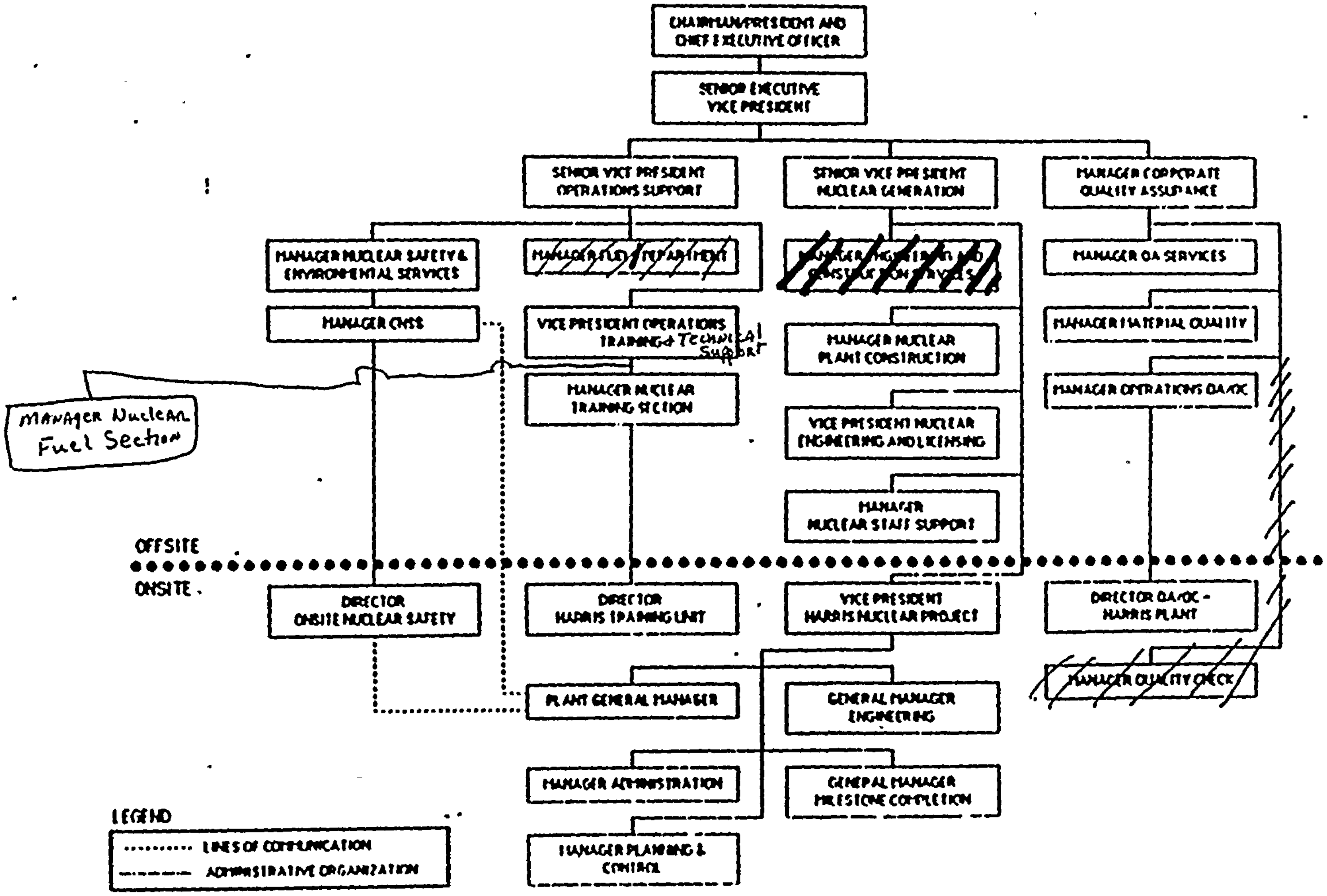

FIGURE 6.2-1

OFFSITE ORGANIZATION

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6.2.3 ONSITE NUCLEAR SAFETY (ONS) UNIT

FUNCTION

(INCLUDING INFORMATION FORWARDED FROM INPO FROM THEIR EVALUATION OF ALL INDUSTRY LER'S)

6.2.3.1 The ONS Unit shall function to examine unit operating characteristics, NRC issuances, industry advisories, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety, to appropriate levels of management, up to and including the Senior Vice President-Operations Support, if necessary.

COMPOSITION

6.2.3.2 The ONS Unit shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a baccalaureate degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ONS Unit shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ONS Unit shall be prepared, maintained, and forwarded each calendar month to the Manager-Nuclear Safety and Environmental Services.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Foreman in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a baccalaureate degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 DELETED ~~UNIT STAFF QUALIFICATIONS~~

~~6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of the September 1979 draft of ANS 3.1, with the exceptions and alternatives noted on FSAR pages 1.8-8 (Am.20), 1.8-9 (Am.17), 1.8-10 (Am.22),~~

*Not responsible for sign-off function.

UNIT STAFF QUALIFICATIONS (Continued)

~~1.8-11 (Am.20), 1.8-12 (Am.17), and 1.8-13 (Am.17), for comparable positions, except for the Manager-Environmental and Radiation Control who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980, NRC letter to all licensees.~~

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director-Harris Training Unit and shall meet or exceed the requirements and recommendations of the September 1979 draft of ANS 3.1, with the exceptions and alternatives noted on FSAR pages 1.8-8 (Am.20), 1.8-9 (Am.17), 1.8-10 (Am.22), 1.8-11 (Am.20), 1.8-12 (Am.17), and 1.8-13 (Am.17), and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 SAFETY AND TECHNICAL REVIEWS

6.5.1.1 General Program Control

6.5.1.1.1 A safety and a technical evaluation shall be prepared for each of the following:

a. All procedures and programs required by Specification 6.8, other procedures that affect nuclear safety, and changes thereto;

b. All proposed tests and experiments that are not described in the Final Safety Analysis Report; and

c. All proposed changes or modifications to plant systems or equipment that affect nuclear safety.

6.5.1.2 Technical Evaluations

6.5.1.2.1 Technical evaluations will be performed by personnel qualified in the subject matter and will determine the technical adequacy and accuracy of the proposed activity. If interdisciplinary evaluations are required to cover the technical-scope of an activity, they will be performed.

6.5.1.2.2 Technical review personnel will be identified by the responsible Manager or his designee for a specific activity when the review process begins.

6.5.1.3 Qualified Safety Reviewers

6.5.1.3.1 The Plant General Manager shall designate those individuals who will be responsible for performing safety reviews described in Specification 6.5.1.4.

Attachment 3 to NLS-86-326

CP&L Comments

SHNPP Final Draft Technical Specifications

Record Number: 706 Comment Type: ERROR

LCO Number: 3.08.04.02 Page Number: 3/4 8-40 - 43

Section Number: TABLE 3.8-2

Comment:

DELETE COLUMN "BYPASS DEVICE" AND PUT A * AFTER THE FUNCTIONAL DESCRIPTION FOR VALVES ON PAGE 8-41 (1AF-55, 1AF-93, 1AF-74, 1AF-137, 1AF-143, 1AF-149, AND 1MS-70); PAGE 8-42 (1MS-72, 3FV-B2, 3FV-B4, 3CZ-B1, 3CZ-B3, 3CZ-B17, 3CZ-B2, 3CZ-B4, 3CZ-B18, AND 3CZ-B14); AND PAGE 8-43 (3CZ-B26, 3CZ-B25, 3CZ-B13, 3CZ-B12, 3CZ-B10, 3CZ-B9, 3CZ-B11, 3CZ-B23, 3CZ-B21, 3CZ-B22, 3CZ-B24, 3CZ-B19, AND 3CZ-B20)

REVISE THE * FOOTNOTE ON PAGE 8-43 TO READ Overload bypass for these valves is accomplished by the activation of slave relays in circuit. These activation slave relays are tested as part of the Engineered Safety Features Actuation System Instrumentation in accordance with the requirements of Table 4.3-2.

Basis

THIS CHANGES REVISES THIS REQUEST IN ACCORDANCE WITH DISCUSSIONS ON 8-14-86 AND 8-28-86 WITH MR. O. CHOPRA OF THE NRR STAFF. AT THAT TIME IT WAS STATED THAT WHILE IT WAS ACCEPTABLE THAT THESE SPECIAL ITEMS ARE TO BE TESTED ELSEWHERE, HE FELT THAT THE BYPASS DEVICE COLUMN SHOULD STILL READ "YES". SINCE THIS WOULD MEAN THAT ALL ITEMS IN THE COLUMN WOULD BE IDENTICAL, CP&L FEELS THAT THE COLUMN CAN BE COMPLETELY DELETED. THE FOOTNOTE HAS BEEN REVISED TO BE AND MORE SPECIFIC ABOUT WHERE THE OTHER TEST REQUIREMENTS MAY BE FOUND.

FINAL DRAFT

TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	~~BYPASS DEVICE (YES/NO)~~
1CS-341 (2CS-V522)	RCP A SEAL ISOL	~~YES~~
1CS-382 (2CS-V523)	RCP B SEAL ISOL	~~YES~~
1CS-423 (2CS-V524)	RCP C SEAL ISOL	~~YES~~
1CS-182 (2CS-V600)	CSIP A MINIFLOW ISOLATION	~~YES~~
1CS-210 (2CS-V601)	CSIP B MINIFLOW ISOLATION	~~YES~~
1CS-196 (2CS-V602)	CSIP C MINIFLOW ISOLATION	~~YES~~
1CS-235 (2CS-V609)	CSIP to RCS ISOLATION	~~YES~~
1CS-166 (2CS-L521)	VCT ISOLATION	~~YES~~
1CS-292 (2CS-L522)	RWST ISOLATION	~~YES~~
1CS-214 (2CS-V585)	CSIPS MINIFLOW ISOLATION	~~YES~~
1CS-165 (2CS-L520)	VCT ISOLATION	~~YES~~
1CS-291 (2CS-L523)	RWST ISOLATION	~~YES~~
1CS-238 (2CS-V610)	CSIP TO RCS ISOLATION	~~YES~~
1CS-170 (2CS-V587)	CSIP SUCTION ISOLATION	~~YES~~
1CS-169 (2CS-V589)	CSIP SUCTION ISOLATION	~~YES~~
1CS-171 (2CS-V590)	CSIP SUCTION ISOLATION	~~YES~~
1CS-168 (2CS-V588)	CSIP SUCTION ISOLATION	~~YES~~
1CS-219 (2CS-V603)	CSIP DISCHARGE ISOL	~~YES~~
1CS-217 (2CS-V604)	CSIP DISCHARGE ISOL	~~YES~~
1CS-218 (2CS-V605)	CSIP DISCHARGE ISOL	~~YES~~
1CS-220 (2CS-V606)	CSIP DISCHARGE ISOL	~~YES~~
1CS-240 (2CS-V611)	SEAL WATER INJECTION	~~YES~~
1CS-278 (2CS-V586)	BORIC ACID TANK TO CSIP	~~YES~~
1CS-746 (2CS-V757)	CSIP MINIFLOW	~~YES~~
1CS-752 (2CS-V759)	CSIP MINIFLOW	~~YES~~
1CS-753 (2CS-V760)	CSIP MINIFLOW	~~YES~~
1CS-745 (2CS-V758)	CSIP MINIFLOW	~~YES~~
1CS-472 (2CS-V517)	RCP~~S~~ SEAL WATER RETURN ISOL	~~YES~~
1CS-470 (2CS-V516)	RCP SEAL WATER ISOLATION	~~YES~~
1RH-25 (2RH-V507)	RHR TO CSIP SUCTION	~~YES~~
1RH-63 (2RH-V506)	RHR TO CSIP SUCTION	~~YES~~
1RH-31 (2RH-F513)	RHR A MINI FLOW	~~YES~~
1RH-69 (2RH-F512)	RHR B MINI FLOW	~~YES~~
1RH-2 (1RH-V503)	RHRS INLET ISOLATION	~~YES~~
1RH-40 (1RH-V501)	RHRS INLET ISOLATION	~~YES~~
1RH-1 (1RH-V502)	RHRS INLET ISOLATION	~~YES~~
1RH-39 (1RH-V500)	RHRS INLET ISOLATION	~~YES~~
1SI-1 (2SI-V503)	BORON INJECTION TANK INLET ISOL	~~YES~~
1SI-4 (2SI-V506)	BORON INJECTION TANK OUTLET ISOL	~~YES~~
1SI-2 (2SI-V504)	BORON INJECTION TANK INLET ISOL	~~YES~~
1SI-3 (2SI-V505)	BORON INJECTION TANK OUTLET ISOL	~~YES~~
1SI-246 (2SI-V537)	ACCUMULATOR A DISCHARGE ISOLATION	~~YES~~
1SI-248 (2SI-V535)	ACCUMULATOR C DISCHARGE ISOLATION	~~YES~~
1SI-300 (2SI-V571)	CNMT SUMP TO RHR PUMP A ISOL	~~YES~~
1SI-310 (2SI-V573)	CNMT SUMP TO RHR PUMP A ISOL	~~YES~~
1SI-247 (2SI-V536)	ACCUM B DISCHARGE ISOLATION	~~YES~~

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	~~BYPASS DEVICE (YES/NO)~~
1SI-301 (2SI-V570)	CNMT SUMP TO RHR PUMP B ISOL	~~YES~~
1SI-311 (2SI-V572)	CNMT SUMP TO RHR PUMP B ISOL	~~YES~~
1SI-107 (2SI-V500)	HH SI TO RCS HL	~~YES~~
1SI-52 (2SI-V502)	HH SI TO RCS CL	~~YES~~
1SI-86 (2SI-V501)	HH SI TO RCS HL	~~YES~~
1SI-326 (2SI-V577)	LH SI TO RCS HL	~~YES~~
1SI-327 (2SI-V576)	LH SI TO RCS HL	~~YES~~
1SI-340 (2SI-V579)	LH SI TO RCS CL	~~YES~~
1SI-341 (2SI-V578)	LH SI TO RCS CL	~~YES~~
1SI-359 (2SI-V587)	LH SI TO RCS HL	~~YES~~
1SI-322 (2SI-V575)	RWST TO RHR A ISOL	~~YES~~
1SI-323 (2SI-V574)	RWST TO RHR B ISOL	~~YES~~
1CC-128 (3CC-B5)	CCS NONESSENTIAL RETURN ISOL	~~YES~~
1CC-127 (3CC-B6)	CCS NONESSENTIAL RETURN ISOL	~~YES~~
1CC-99 (3CC-B19)	CCS NONESSENTIAL RETURN ISOL	~~YES~~
1CC-113 (3CC-B20)	CCS NONESSENTIAL RETURN ISOL	~~YES~~
1CC-147 (3CC-V165)	RHR COOLING ISOL	~~YES~~
1CC-167 (3CC-V167)	RHR COOLING ISOL	~~YES~~
1CC-176 (2CC-V172)	CVCS HX CNMT ISOLATION	~~YES~~
1CC-202 (2CC-V182)	CVCS HX CNMT ISOLATION	~~YES~~
1CC-208 (2CC-V170)	CCW-RCPS ISOLATION	~~YES~~
1CC-299 (2CC-V183)	RCPS BEARING HX ISOLATION	~~YES~~
1CC-251 (2CC-V190)	RCPS THER BARRIER ISOLATION	~~YES~~
1CC-207 (2CC-V169)	CCW-RCPS ISOLATION	~~YES~~
1CC-297 (2CC-V184)	RCPS BEARING HX ISOLATION	~~YES~~
1CC-249 (2CC-V191)	RCPS THER BARRIER ISOLATION	~~YES~~
1CT-105 (2CT-V6)	CNMT SPRAY SUMP A RECIRC ISOL	~~YES~~
1CT-102 (2CT-V7)	CNMT SPRAY SUMP B RECIRC ISOL	~~YES~~
1CT-26 (2CT-V2)	CNMT SPRAY PUMP A INJECT. SUPPLY	~~YES~~
1CT-71 (2CT-V3)	CNMT SPRAY PUMP B INJECT. SUPPLY	~~YES~~
1CT-50 (2CT-V21)	SPRAY HDR A ISOLATION	~~YES~~
1CT-12 (3CT-V85)	NAOH ADDITIVE ISOLATION	~~YES~~
ICT-88 (2CT-V43)	SPRAY HDR B ISOLATION	~~YES~~
ICT-11 (3CT-V88)	NAOH ADDITIVE ISOLATION	~~YES~~
1CT-47 (2CT-V25)	CNMT SPRAY HDR A RECIRC	~~YES~~
1CT-24 (2CT-V8)	CNMT SPRAY PUMP A EDUCTOR TEST	~~YES~~
1CT-95 (2CT-V49)	CNMT SPRAY HDR B RECIRC	~~YES~~
1CT-25 (2CT-V145)	CNMT SPRAY PUMP B EDUCTOR TEST	~~YES~~
1AF-5 (3AF-V187)	AFWP A RECIRC	~~YES~~
1AF-24 (3AF-V188)	AFWP B RECIRC	~~YES~~
1AF-55 (2AF-V10)	AFW TO SG A ISOL *	~~YES~~
1AF-93 (2AF-V19)	AFW TO SG B ISOL *	~~YES~~
1AF-74 (2AF-V23)	AFW TO SG C ISOL *	~~YES~~
1AF-137 (2AF-V116)	AFWTD TO SG A ISOL *	~~YES~~
1AF-143 (2AF-V117)	AFWTD TO SG B ISOL *	~~YES~~
1AF-149 (2AF-V118)	AFWTD TO SG C ISOL *	~~YES~~
1MS-70 (2MS-V8)	AFWTD STEAM B ISOLATION *	~~YES~~

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	~~BYPASS DEVICE (YES/NO)~~
1MS-72 (2MS-V9)	AFWTD STEAM C ISOLATION *	~~YES~~
1SW-39 (3SW-B5)	NORMAL SW HDR A ISOLATION	~~YES~~
1SW-276 (3SW-B8)	NORMAL SW HDR A RETURN ISOL	~~YES~~
1SW-270 (3SW-B15)	SW HDR A TO AUX RSVR ISOL	~~YES~~
1SW-40 (3SW-B6)	NORMAL SW HDR B ISOL	~~YES~~
1SW-275 (3SW-B13)	SW HDR A RETURN ISOL	~~YES~~
1SW-274 (3SW-B14)	SW HDR B RETURN ISOL	~~YES~~
1SW-271 (3SW-B16)	SW HDR B TO AUX RSVR ISOL	~~YES~~
~~1SW-3 (3SW-B3)~~	~~EMER SW PUMP 1A MAIN RSVR INLET~~	~~YES~~
~~1SW-4 (3SW-B4)~~	~~EMER SW PUMP 1B MAIN RSVR INLET~~	~~YES~~
~~1SW-1 (3SW-B1)~~	~~EMER SW PUMP 1A AUX RSVR INLET~~	~~YES~~
~~1SW-2 (3SW-B2)~~	~~EMER SW PUMP 1B AUX RSVR INLET~~	~~YES~~
1SW-92 (2SW-B46)	SW TO FAN CLR AH3 INLET	~~YES~~
1SW-97 (2SW-B47)	SW FAN CLR AH3 OUTLET	~~YES~~
1SW-91 (2SW-B45)	SW TO FAN CLR AH2 INLET	~~YES~~
1SW-109 (2SW-B49)	FROM SW FAN CLR AH2 OUTLET	~~YES~~
1SW-225 (2SW-B52)	SW TO FAN CLR AH1 INLET	~~YES~~
1SW-98 (2SW-B48)	SW FAN CLR AH1 OUTLET	~~YES~~
1SW-227 (2SW-B51)	SW TO FAN CLR AH4 INLET	~~YES~~
1SW-110 (2SW-B50)	SW FAN CLR AH4 OUTLET	~~YES~~
1SW-124 (3SW-B70)	SW TO AFWTD PUMP	~~YES~~
1SW-126 (3SW-B71)	SW TO AFWTD PUMP	~~YES~~
1SW-129 (3SW-B73)	SW TO AFWTD PUMP	~~YES~~
1SW-127 (3SW-B72)	SW TO AFWTD PUMP	~~YES~~
1SW-123 (3SW-B75)	SW TO AFW PUMP A SUPPLY	~~YES~~
1SW-121 (3SW-B74)	SW TO AFW PUMP A SUPPLY	~~YES~~
1SW-132 (3SW-B77)	SW TO AFW PUMP B SUPPLY	~~YES~~
1SW-130 (3SW-B76)	SW TO AFW PUMP B SUPPLY	~~YES~~
1ED-94 (2MD-V36)	CNMT SUMP ISOLATION	~~YES~~
1ED-95 (2MD-V77)	CNMT SUMP ISOLATION	~~YES~~
3CZ-B5	RAB ELEC PROT INLET	~~YES~~
3CZ-B6	RAB ELEC PROT INLET	~~YES~~
3CZ-B7	RAB ELEC PROT EXHAUST	~~YES~~
3CZ-B8	RAB ELEC PROT EXHAUST	~~YES~~
3CZ-B32	RAB ELEC PROT PURGE MAKE-UP	~~YES~~
3CZ-B33	RAB ELEC PROT PURGE MAKE-UP	~~YES~~
3CZ-B34	RAB ELEC PROT PURGE INLET	~~YES~~
3CZ-B35	RAB ELEC PROT PURGE INLET	~~YES~~
3FV-B2	FUEL HANDLING EXHAUST INLET *	~~NO*~~
3FV-B4	FUEL HANDLING EXHAUST INLET *	~~NO*~~
3CZ-B1	CONTROL ROOM NORMAL SUPPLY ISOL *	~~NO*~~
3CZ-B3	CONTROL ROOM NORMAL EXHAUST ISOL *	~~NO*~~
3CZ-B17	CONTROL ROOM PURGE MAKE UP *	~~NO*~~
3CZ-B2	CONTROL ROOM NORMAL SUPPLY ISOL *	~~NO*~~
3CZ-B4	CONTROL ROOM EXHAUST ISOLATION *	~~NO*~~
3CZ-B18	CONTROL ROOM PURGE MAKE UP *	~~NO*~~
3CZ-B14	CONTROL ROOM PURGE EXHAUST *	~~NO*~~

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION	~~BYPASS DEVICE (YES/NO)~~
3CZ-B26	CONTROL ROOM NORMAL SUPPLY DISCH *	~~NO~~*
3CZ-B25	CONTROL ROOM SUPPLY DISCHARGE *	~~NO~~*
3CZ-B13	CONTROL ROOM PURGE EXHAUST *	~~NO~~*
3CZ-B12	CNTL RM EMER FLTR OUTSIDE AIR INTAKE *	~~NO~~*
3CZ-B10	CNTL RM EMER FLTR OUTSIDE AIR INTAKE *	~~NO~~*
3CZ-B9	CNTL RM EMER FLTR OUTSIDE AIR INTAKE *	~~NO~~*
3CZ-B11	CNTL RM EMER FLTR OUTSIDE AIR INTAKE *	~~NO~~*
3CZ-B23	CONTROL ROOM EMER FLTR INLET *	~~NO~~*
3CZ-B21	CONTROL ROOM FLTR DISCHARGE *	~~NO~~*
3CZ-B22	CONTROL ROOM EMER FLTR DISCHARGE *	~~NO~~*
3CZ-B24	CONTROL ROOM EMER FLTR INLET *	~~NO~~*
3CZ-B19	CONTROL ROOM EMER FLTR DISCHARGE *	~~NO~~*
3CZ-B20	CONTROL ROOM EMER FLTR DISCHARGE *	~~NO~~*
3AV-B1	RAB EMER EXHAUST INLET	~~YES~~
3AV-B2	RAB EMER EXHAUST OUTLET	~~YES~~
3AV-B4	RAB EMER EXHAUST INLET	~~YES~~
3AV-B5	RAB EMER EXHAUST OUTLET	~~YES~~
3AV-B3	RAB EMER EXHAUST BLEED	~~YES~~
3AV-B6	RAB EMER EXHAUST BLEED	~~YES~~
3AC-B2	RAB SWGR B EXHAUST	~~YES~~
3AC-B3	RAB SWGR B EXHAUST	~~YES~~
3AC-B1	RAB SWGR A EXHAUST	~~YES~~

*~~Included for completeness only~~. Overload bypass for these valves is accomplished by the Activation ~~circuit design and no device is present~~ slave Relays in the circuit. These activation slave relays are tested as a part of the Engineered Safety Features Actuation System Instrumentation in accordance with the the Requirements of Table 4.3-2.

CP&L Comments

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Record Number: 727 Comment Type: ERROR

LCO Number: 3.03.03.05.a Page Number: 3/4 3-64

Section Number: TABLE 3.3-9

Comment:

CHANGE TABLE 3.3-9 TO THE ATTACHED MARKUP.

Basis

THE CHANGE FOR ITEMS 1 AND 2 IS TO BE CONSISTANT WITH THE GUIDANCE OF REG GUIDE 0800 WHICH INDICATES THAT THE MINIMUM ACCEPTABLE CHANNELS IS ONE HOT LEG AND ONE COLD LEG TEMPERATURE INDICATION FOR EACH STEAM GENERATOR REQUIRED TO BE OPERABLE. AT SHNPP THIS NUMBER OF STEAM GENERATORS IS TWO.
ON ITEM 4, THE PHRASE "-SSA CHANNEL" HAS BEEN ADDED TO CLARIFY THAT OUR ANALYSIS ASSUMED SPECIFIC CHANNELS BE OPERABLE AND THUS THIS CHANGE AND MORE RESTICTIVE THAN THE PERVIOUS ENTRY.
THE LOCATION OF "NOTE 2" ON ITEM 7 IS MOVED FOR GREATER CLARITY. THE NECESSARY MINIMUM CHANNEL IS THE B RHR TRAIN AND PLACING THE NOTE IN THAT COLUMN IS A MORE LOGICAL POAITION THAN ITS CURRENT LOCATION.

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant System Hot-Leg Temperature	ACP*	2	~~1~~ 2
2. Reactor Coolant System Cold-Leg Temperature	ACP*	2	~~1~~ 2
3. Pressurizer Pressure	ACP*	2	1-SSA Channel**
4. Pressurizer Level	ACP*	2	1-SSA CHANNEL**
5. Steam Generator Pressure (Note 1)	ACP*	1/Steam Generator	1/Steam Generator
6. Steam Generator Water Level--Wide Range (Note 1)	ACP*	1/Steam Generator	1/Steam Generator
7. Residual Heat Removal Flow ~~(Note 2)~~	ACP*	2	1 (Note 2)
8. Auxiliary Feedwater Flow (Note 1)	ACP*	1/Steam Generator	N.A. (Note 3)
9. Condensate Storage Tank Level	ACP*	2	1-SSA Channel**
10. Reactor Coolant System Pressure-Wide Range	ACP*	2	1-SSA Channel**
11. Wide-Range Flux Monitor (SR INDICATOR)	ACP*	1	1-SSA Channel**
12. Charging Header Flow	ACP*	1	1-SSA Channel**
13. a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP or	ACP*	1	1-SSA Channel**
b. Auxiliary Feedwater Turbine Speed	ACP*	1	1-SSA Channel**
14. Boric Acid Tank Level	ACP*	1	1-SSA Channel**

*ACP = Auxiliary Control Panel
**SSA = Safe Shutdown Analysis

Note 1 - Steam Generators A&B Only
Note 2 - RHR Train B Only
Note 3 - Steam Generator Water Level is used

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CP&L Comments

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Record Number: 737 Comment Type: ERROR

LCO Number: 5.07.01 Page Number: 5-8

Section Number: TABLE 5.7-1

Comment:

IN THE DESIGN CYCLE OR TRANSIENT COLUMN FOR THE REACTOR COOLANT SYSTEM 10 AUXILIARY SPRAY ACTUATION CYCLES, CHANGE ",625 F" TO "Greater than 320 F but less than 625 F."

Basis

THIS CHANGE IS REQUIRED TO MAKE THE SPECS MORE ACCURATE. THE CYCLE IS FOR THE TEMPERATURE RANGE >320 F TO <625 F. ACTUATION BELOW 320 F DOES NOT APPLY TO THIS CYCLIC LIMIT.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to > 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	200 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential ~~$\leq$ 625°F.~~ greater than 320°F but less than 625°F.
	200 leak tests.	Pressurized to $\geq$ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 1481 psig.

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CP&L Comments

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Record Number: 739 Comment Type: ERROR

LCO Number: 3.08.01.01 Page Number: 3/4 8-7

Section Number: 4.8.1.1.2.f.10

Comment:

ADD A NEW ITEM 4.8.1.1.2.f.10.f AS FOLLOWS:
f. Loss of generator potential transformer circuit.
ADD TO ITEM 4.8.1.1.2.f.6.C "loss of generator potential transformer circuit" AS AN ADDITIONAL SIGNAL.

Basis

THE ADDITION OF THIS ITEM IS NECESSARY TO ENSURE THAT ALL DIESEL LOCKOUT FEATURES ARE TESTED.

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SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected shutdown loads through the load sequencer, and operating for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz.

5. Verifying that on a safety injection test signal (without loss of power) the diesel generator starts** on the auto-start signal and operates on standby for greater than or equal to 5 minutes.

6. Simulating a loss of offsite power in conjunction with a safety injection test signal, and

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected emergency (accident) loads through the sequencing timers, and operating for greater than or equal to 5 minutes and maintaining the steady-state voltage and frequency at 6900 ± 690 volts and 60 ± 1.2 Hz.

LOSS OF GENERATOR POTENTIAL TRANSFORMER CIRCUIT

c) Verifying that all diesel generator trips, except engine overspeed, generator differential, and emergency bus differential are automatically bypassed upon loss of offsite power signal a safety injection signal.

IN CONJUNCTION WITH

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

SURVEILLANCE REQUIREMENTS (Continued)

7. Verifying the diesel generator operates** for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 6800-7000 kW*** and, during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6200-6400 kW***. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.f.6 b).#

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 6500 kW;

9. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Proceed through its shutdown sequence.

10. Verifying that the following diesel generator lockout features prevent diesel generator operation:

    a) Engine overspeed
    b) Generator differential
    c) Emergency bus differential
    d) Emergency Stop
    e) Operational and maintenance switch in the maintenance mode.
    f) LOSS OF GENERATOR POTENTIAL TRANSFORMER CIRCUIT

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 450 rpm in less than or equal to 10 seconds.

h. At least once per 10 years by:

   1) Draining each main fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or other appropriate cleaning solution, and

---

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

#If Specification 4.8.1.1.2f.6 b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 6200-6400 kW for 1 hour or until operating temperature has stabilized.

CP&L Comments

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Record Number: 756 Comment Type: ERROR

LCO Number: B 2.02.01 Page Number: B 2-5,6 & 7

Section Number:. B 2.2.1

Comment:

ON PAGE B 2-5 & 6 CHANGE THE DESCRIPTION OF P-7 TO "...RATED THERMAL POWER or turbine impulse...."

ON PAGE B 2-7 IN THE PARAGRAPH ON TURBINE TRIP, INSERT "or a turbine impulse pressure at approximately 10% of full power equivalent" AFTER "RATED THERMAL POWER".
ON PAGE B 2-5, 6 & 7 IN THE PARAGRAPHS ON PRESSURIZER PRESSURE, PRESSURIZER WATER LEVEL, UNDERVOLTAGE AND UNDERFRENQUENCY, AND TURBINE TRIP, CHANGE THE PHRASE "blocked by P-7" TO "blocked by the loss of P-7".

Basis

THIS CHANGE IS TO PROVIDE A MORE COMPLETE DESCRIPTION OF THE P-7 INTERLOCK CONSISTENT WITH OTHER BASES DISCUSSIONS. THIS IS NECESSARY TO CLARIFY WHETHER THE PRESENCE OR ABSENCE OF THE P-7 SIGNAL PROVIDES FOR THE ACTIONS DISCUSSED.

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Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER ~~with~~ OR turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER ~~with~~ OR a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 91.7% of nominal full loop flow. Above P-8

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 91.7% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.627×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 38.5%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and the loss of Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER ~~with~~ OR a turbine impulse chamber pressure at approximately

BASES

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing the loss of power the Reactor trip from the Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

CP&L Comments

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Record Number: 757 | Comment Type: ERROR

LCO Number: 3.07.12 | Page Number: 3/4 7-41

Section Number: TABLE 3.7-6

Comment:

REVISE TABLE 3.7-6 PER THE ATTACHED MARKUP.

Basis

THE ELEVATION CHANGES ARE MADE TO PROVIDE ADDITIONAL INFORMATION OVERLOOKED ON PRIOR SUBMITTALS AND TO CORRECT TYPOGRAPHICAL ERRORS IN PREVIOUS CP&L SUBMITTALS.
ITEM 23 IS DELETED BECAUSE THE EXHAUST SILENCERS PERFORM NO SAFETY FUNCTION AND THEREFORE DO NOT NEED TO HAVE A TEMPERATURE LIMIT SPECIFICIED.
THE CHANGES TO ITEM 19 ARE TO CORRECT AN ERROR IN THE ELEVATION AND TO PREVENT POSSIBLE CONFUSION IN THE WORDING. THE EQUIPMENT OF INTEREST HERE IS NOT THE CONDENSATE STORAGE TANK ITSELF, BUT TRANSMITTERS WHICH ARE LOCATED IN THE TANK AREA BELOW THE TANK.

TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA		MAXIMUM TEMPERATURE LIMIT (°F)
REACTOR AUXILIARY BUILDING		
1.	Control Room Envelope, (El 305')	85
2.	Process I&C, Room (El 305')	85
3.	Rod Control Cabinets Area (El 305')	104
4.	A&B Battery Rooms (El 286')	85
5.	A&B Switchgear Rooms (El 286')	90
6.	Main Steam, Feedwater Pipe Tunnel (El 286' & 261')	116
7.	SA&SB Electrical Penetration Areas (El ~~236')~~ 261 & 286	104
8.	Area with MCC 1A35~~/~~SA and 1B35SB (El 261')	104
9.	HVAC Chillers, Auxiliary FW Piping & Valve Area (El ~~236')~~ 261	104
10.	CCW Pumps, CCW Hx, Auxiliary FW Pumps Area (El 236')	104
11.	1A-SA, 1B-SB, and 1C-SAB ~~Spare~~ Charging Pump Rooms (El 236')	104
12.	Service Water Booster Pump 1B-SB (El 236')	104
13.	Mechanical and Electrical Penetration Areas (El 236')	104
14.	Containment Spray Additive Tank, and H&V Equipment Area (El 216')	104
15.	Trains A&B Containment Spray Pump, RHR Pump, H&V Equipment Areas (El 190')	104
FUEL HANDLING BUILDING		
16.	Trains A&B Emergency Exhaust System Areas (El 261')	104
17.	Fuel Pool Cooling Pump and Heat Exchanger Area (El 236')	104
WASTE PROCESSING BUILDING		
18.	H&V Equipment Room (El 236')	104
MISCELLANEOUS		
19.	~~Condensate Storage~~ Tank Area (El ~~261~~ 236')	122
20.	Diesel Fuel Oil Storage Building (El 242')	122
21.	Emergency Service Water Electrical Equipment Room	104
22.	Emergency Service Water Pump Room	122
23.	~~1A-SA & 1B-SB Exhaust Silencer Rooms (El 292')~~	~~122~~ DELETED
24.	1A-SA & 1B-SB H&V Equipment Rooms (El 292')	122
25.	1A-SA & 1B-SB H&V Equipment Rooms (El 280')	~~110~~ 118
26.	1A-SA & 1B-SB Electrical Rooms (El 261')	~~104~~ 116
27.	1A-SA & 1B-SB Diesel Generator Rooms (El 261')	120

CP&L Comments
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Record Number: 761 | Comment Type: ERROR

LCO Number: 3.03.02 | Page Number: 3/4 3-34

Section Number: TABLE 3.3-4

Comment:

REVISE ITEM 9 OF TABLE 3.3-4 TO THE ATTACHED MARKUP.

Basis

THIS CHANGE IS MADE TO RETURN THE SPECIFICATION TO THE STANDARD FORMAT. THE DELETED DATA IS, AND SHOULD BE, PROCEDURALLY CONTROLED. THE DELETED DATA IS NOT RELATED TO ANY SAFETY ANALYSIS ASSUMPTION AND THEREFORE SHOULD NOT BE IN THE TECHNICAL SPECIFICATIONS.

THE CHANGE TO LESS THAN OR EQUAL TO 16 sec IN ITEM 9b IS TO COVER THE ANALYTICAL ASSUMPTIONS IN A MORE CONSERVATIVE MANNER BY SLIGHTLY REDUCEING THE TIME DELAY PRIOR TO THE DIESEL GENERATOR START.

CHANGES TO THE SETPOINTS ARE TO PROVIDE THE LATEST INFORMATION AVAILABLE TO US FROM THE A/E.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE ~~DROPOUT~~	~~PICKUP~~
9. Loss-of-Offsite Power						
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	≥ 4830 volts with a ≤ 1.0 second time delay.	~~≥ 4900 volts and~~ ≥ 4692 volts with a time delay ~~be- tween 0.95 and~~ ≤1.5 seconds.	~~N.A.~~
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	≥ ~~6424~~ 6420 volts with a ~~≤ 15.6~~ 16 second time delay (with Safety Injection).	≥ ~~6400~~ 6392 volts with a time delay ~~between 14 and~~ ≤ 18 seconds (with Safety Injection).	~~≤ 6478 volts~~
				≥ ~~6424~~ 6420 volts with a ≤ 54.0 second time delay (with- out Safety Injection).	≥ ~~6400~~ 6392 volts with a ≤ 60 second time delay (with- out Safety Injection).	~~≤ 6478 volts~~
10. Engineered Safety Features Actuation System Interlocks						
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≥ 2000 psig	≥ 1986 psig	
Not P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2014 psig	
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 550.6°F	

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CP&L Comments

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Record Number: 770 | Comment Type: IMPROVEMENT

LCO Number: 3.06.02.01 | Page Number: 3/4 6-11

Section Number: 4.6.2.1.b

Comment:

CHANGE THE SURVEILLANCE TO THE FOLLOWING:
By verifying, that on an indicated recirculation flow of at least 2150 gpm, each pump develops a discharge pressure of greater than or equal to 229 psig when tested pursuant to Specification 4.0.5.

Basis

THE CURRENT TECH SPEC VALUE OF 245 psig DISCHARGE PRESSURE HAS, AS ITS BASIS, AN UNWRITTEN ASSUMPTION OF AN INDICATED FLOW RATE OF APPROXIMATELY 1500-1550 gpm. IT MERELY REPRESENTS A POINT CHOSEN ON THE PUMP CURVE AS A PLACE TO REPEATABLY MEASURE PUMP PERFORMANCE. THE ACTUAL FLOW RATE OF THE CONTAINMENT SPRAY PUMPS AND THEIR DISCHARGE PRESSURE WILL VARY CONSIDERABLY THROUGH THE COURSE OF AN ACCIDENT DUE TO THE DROP IN RWST LEVEL AND SUBSEQUENT DROP IN SUCTION HEAD.
THE DIFFICULTY WITH LEAVING THIS VALUE IN THE TECH SPEC'S IS THAT IT CANNOT BE READILY TESTED. THE ONLY MECHANISM AVAILABLE TO US TO THROTTLE FLOW IS THROUGH A NORMALLY LOCKED OPEN VALUE. MOVING THIS VALVE ROUTINELY CREATES AN UNDESIRABLE AND UNNECESSARY RISK OF PERSONNEL ERROR OF LEAVING THE VALVE MISPOSITIONED. WHAT CP&L HAS CHOSEN TO DO INSTEAD IS TO CHANGE THE TECH SPEC TO USE AS A REFERENCE POINT A SET OF PARAMENTERS WHICH REPRESENT THE SYSTEM AS IT WOULD NORMALLY BE RUN FOR THE TEST WITHOUT CHANGING THE POSITION OF LOCKED VALVES. WE HAVE ALSO SPECIFIED A MINIMUM INDICATED FLOW IN ORDER TO BE MORE DEFINITIVE. THE NEW DATA POINT REPRESENTS ONLY ANOTHER POINT ON THE SAME PUMP CURVE AS BEFORE AND THERFORE PROVIDES THE SAME MARGINS AS BEFORE BUT WITHOUT UNNECESSARY RISKS AND PLANT CHANGES.

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AUG 1986

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on AN INDICATED recirculation flow OF AT LEAST 2150 gpm, each pump develops a discharge pressure of greater than or equal to ~~245~~ 229 psig when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal and

2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

3. Verifying that COINCIDENT WITH AN INDICATION OF CONTAINMENT SPRAY PUMP RUNNING, each automatic ~~recirculation~~ valve from the sump AND RWST actuates TO ITS APPROPRIATE POSITION FOLLOWING on an RWST Lo-Lo test signal ~~and safety injection test signal.~~

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CP&L Comments

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Record Number: 773 | Comment Type: IMPROVEMENT

LCO Number: 3.03.02 | Page Number: 3/4 3-41, 44&45

Section Number: TABLE 4.3-2

Comment:

ITEMS 1b, 1c, 4a AND 4b - UNDER THE "MODES FOR WHICH SURVEILLANCE IS REQUIRED" COLUMN, ADD MODE 4.
ITEM 4e - UNDER THE "MODE FOR WHICH SURVEILLANCE IS REQUIRED" COLUMN, CHANGE TO 3**, 4**.
UNDER THE TABLE NOTATIONS, ADD NEW NOTE - ** Trip function automatically blocked above P-11 and may be blocked below P-11 when safety injection or low steam lime pressure is not blocked.

Basis

THESE CHANGES ARE NEEDED FOR CONSISTENCY WITH MODE CHANGES TO ITEMS 1b, 1c, 4a AND 4e OF TABLE 3.3-3 WHICH WERE REQUESTED BY NRR REVIEWS IN APRIL 1986 AND AGREED TO BY CP&L. THE NOTE SIMPLY AGREES WITH THE PREVIOUS TABLE ON THESE MODES. THIS RECORD WAS REVISED IN AUGUST BY CP&L TO CORRECT A MARKUP ERROR IN WHICH TWO OF THE NEEDED CORRECTIONS WERE MOT MARKED.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure--High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
(2) Preentry Purge Detector	See Table 4.3-3, Item 1b2, for surveillance requirements.							
c) Airborne Particulate Radioactivity								
(1) RCS Leak Detection (normal purge)	See Table 4.3-3, Item 1C1, for surveillance requirements.							
(2) Preentry Purge Detector	See Table 4.3-3, Item 1C2, for surveillance requirements.							
5) Manual Phase A Isolation	See Item 3.a.1) above for Manual Phase A Isolation Surveillance Requirements.							
4. Main Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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PRELIMINARY

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Main Steam Line Isolation (Continued)								
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Surveillance Requirements.							
e. Negative Steam Line Pressure Rate--High	S	R	M	N.A.	N.A.	N.A.	N.A.	3**, 4**
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level--High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
d. Safety Injection Start Motor-Driven Pump	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Power Surveillance Requirements.							

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BASES

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K (P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.099;
K_2	=	0.0182/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 20 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 \left[T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.086,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

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2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer, and the RCS piping, pumps, valves and fittings are designed to Section III, Division I of the ASME Code for Nuclear Power Plants, which permits a maximum transient pressure of 110% to 125% of design pressure (2485 psig) depending on component. The Safety Limit of 2735 psig (110% of design pressure) is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. For example, if a bistable has a trip setpoint of 100%, a span of 125%, and a calibration accuracy of 0.5% of span, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is $\leq$ 100.62%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of

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BASES

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 91.7% of nominal full loop flow. Above P-8

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 91.7% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.627×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 38.5%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure

LIMITING SAFETY SYSTEM SETTINGS

BASES

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1770 pcm for 3-loop operation.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1770 pcm, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1770 pcm:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. Within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,
2) Control rod position,
3) Reactor Coolant System average temperature,
4) Fuel burnup based on gross thermal energy generation,
5) Xenon concentration, and
6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ±1000 pcm at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If later experience shows adjustment is desirable at approximately 60 EFPD, the adjustment is permissible.

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 2000 pcm.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 2000 pcm immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 2000 pcm:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,
2) Control rod position,
3) Reactor Coolant System average temperature,
4) Fuel burnup based on gross thermal energy generation,
5) Xenon concentration, and
6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER; and

b. Less negative than -42 pcm/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only**.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the above limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

---

*With k_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2000 pcm at 200°F within the next 6 hours; restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging/safety injection pumps shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the Reactor Coolant System and reactor coolant pump seals, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to Specification 4.0.5.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid tank with:

1. A minimum contained borated water volume of 4900 gallons, which is equivalent to 10% indicated level,

2. A boron concentration of between 7000 and 7750 ppm, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 82,000 gallons, which is equivalent to 6% indicated level,

2. A boron concentration of between 2000 and 2200 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

3. Verifying the boric acid tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. Measured RCS flow rate $\geq$ 292,800 gpm x $(1.0 + C_1)$, and

b. Measured $F^N_{\Delta H} \leq 1.49\ [1.0 + 0.2(1.0-P)]$

Where:

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map, and the measured values of $F^N_{\Delta H}$ shall be used for comparison above since the 1.49 value above accounts for a 4% allowance on incore measurement of $F^N_{\Delta H}$.

C_1 = Measurement uncertainty for core flow as described in the Bases.

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F^N_{\Delta H}$ outside the above limits:

a. Within 2 hours either:

1. Restore RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature ΔT	3	2	2	1, 2	6#
8. Overpower ΔT	3	2	2	1, 2	6#
9. Pressurizer Pressure--Low (Above P-7)	3	2	2	1	6#(1)
10. Pressurizer Pressure--High	3	2	2	1, 2	6#
11. Pressurizer Water Level--High (Above P-7)	3	2	2	1	6#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. each operating stm. gen.	1, 2	6#(1)
14. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feedwater flow mismatch in each stm. gen.	1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feedwater flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6#
15. Undervoltage--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Underfrequency--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6#
17. Turbine Trip (Above P-7)					
a. Low Fluid Oil Pressure	3	2	2	1	6#
b. Turbine Throttle Valve Closure	4	4	1	1	10#
18. Safety Injection Input from ESF	2	1	2	1, 2	8
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1	7
or					
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	7

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1)The applicable MODES and ACTION Statement for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, and verify compliance with the shutdown margin requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

b. With no channels OPERABLE, open the Reactor Trip System breakers within 1 hour and suspend all operations involving positive reactivity changes. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	15*
d. Pressurizer Pressure--Low	3	2	2	1, 2, 3#	15*
e. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15*

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Main Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	21
c. Containment Pressure--High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low initiating functions and requirements.				
e. Negative Steam Line Pressure Rate--High	3/steam line	2 in any steam line	2/steam line	3***, 4***	15*
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2	19*
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Steam Generator Water Level--Low-Low					
1) Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. in any stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
2) Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. in any 2 stm. gen.	2/stm. gen. in each stm. gen.	1, 2, 3	15*
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for Loss of Offsite Power initiating functions and requirements.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	1/pump	1/pump	1/pump	1, 2	15*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	3/steam line	2/steam line twice with any steamline low	2/steam line	1, 2, 3	15*
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for all Steam Line Isolation initiating functions and requirements				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	16
Coincident With Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Containment Spray Switch-over to Containment Sump (Continued)					
b. RWST--Low Low	See Item 7.b. above for all RWST--Low Low initiating functions and requirements.				
Coincident With Containment Spray	See Item 2 above for all Containment Spray initiating functions and requirements.				
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus--Undervoltage Primary	3/bus	2/bus	2/bus	1, 2, 3, 4	15*
b. 6.9 kV Emergency Bus--Undervoltage Secondary	3/bus	2/bus	2/bus	1, 2, 3, 4	15*
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure,					
P-11	3	2	2	1, 2, 3	20
Not P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	3	2	2	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Water Level, P-14	See Item 5.b. above for all P-14 initiating functions and requirements.				

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated equipment inoperable and take the appropriate ACTION required in accordance with the specific equipment specification.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - During CORE ALTERATIONS or movement of irradiated fuel within containment, comply with the ACTION statement of Specification 3.9.9.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-1	2.7	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Pressurizer Pressure--Low	18.8	14.41	1.5	≥ 1850 psig	≥ 1836 psig
e. Steam Line Pressure--Low	17.7	14.81	1.5	≥ 601 psig	≥ 578.3 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	3.6	0.71	1.5	≤ 10.0 psig	≤ 11.0 psig

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Main Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	2.7	0.71	1.5	≤3.0 psig	≤3.6 psig
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Trip Setpoints and Allowable Values.				
e. Negative Steam Line Pressure Rate--High	2.3	0.5	0	≤ 100 psi#	≤ 122.8 psi#**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	5.0	1.47	3.0	≤ 100psi	≤ 127.4 psi
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for Main Steam Line Isolation Trip Setpoints and Allowable Values.				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	N.A.	N.A.	N.A.	≥ 23.4%	≥ 20.4%
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST--Low-Low	See Item 7.b. above for all RWST--Low-Low Trip Setpoints and Allowable Values.				
Coincident With Containment Spray	See Item 2. above for all Containment Spray Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	≥ 4830 volts with a ≤ 1.0 second time delay.	≥ 4692 volts with a time delay ≤ 1.5 seconds
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	≥ 6420 volts with a < 16 second time delay (with Safety Injection).	≥ 6392 volts with a time delay ≤ 18 seconds (with Safety Injection).
				≥ 6420 volts with a ≤ 54.0 second time delay (without Safety Injection).	≥ 6392 volts with a ≤ 60 second time delay (without Safety Injection).
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure,					
P-11	N.A.	N.A.	N.A.	≥ 2000 psig	≥ 1986 psig
Not P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2014 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 550.6°F

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \geq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate--High is less than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

#The indicated values are the effective, cumulative, rate-compensated pressure drops as seen by the comparator.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Containment Ventilation Isolation	N.A.
e. Steam Line Isolation	N.A.
f. Reactor Trip	N.A.
g. Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Containment Phase "A" Isolation	$< 62^{(2)}/72^{(1)}$
4) Containment Ventilation Isolation	$\leq 4.75^{(6)}$
5) Auxiliary Feedwater Motor-Driven Pumps	≤ 60
6) Emergency Service Water Pumps	$\leq 32^{(1)}/22^{(8)}$
7) Containment Fan Coolers	$\leq 27^{(1)}/17^{(8)}$
8) Control Room Isolation	N.A.
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Containment Phase "A" Isolation	$< 62^{(2)}/72^{(1)}$
4) Containment Ventilation Isolation	$\leq 4.75^{(6)}$

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low (Continued)	
a. Safety Injection (ECCS) (Continued)	
5) Auxiliary Feedwater Motor-Driven Pumps	$\leq$ 60
6) Emergency Service Water Pumps	$\leq 32^{(1)}/22^{(8)}$
7) Containment Fan Coolers	$\leq 27^{(1)}/17^{(8)}$
8) Control Room Isolation	N.A.
4. Main Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 12^{(5)}/22^{(4)}$
1) Reactor Trip	$\leq$ 2
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Containment Phase "A" Isolation	$< 62^{(2)}/72^{(1)}$
4) Containment Ventilation Isolation	$\leq 4.75^{(6)}$
5) Auxiliary Feedwater Motor-Driven Pumps	$\leq$ 60
6) Emergency Service Water Pumps	$\leq 32^{(1)}/22^{(8)}$
7) Containment Fan Coolers	$\leq 27^{(1)}/17^{(8)}$
8) Control Room Isolation	N.A.
b. Steam Line Isolation	$\leq 12^{(9)}$
5. Containment Pressure--High-3	
a. Containment Spray	$\leq 18.5^{(8)}/32.2^{(1)}$
b. Phase "B" Isolation	$\leq 22.5^{(1)}/12^{(2)}$
6. Containment Pressure--High-2 Steam Line Isolation	$\leq 12^{(9)}$
7. Negative Steam Line Pressure Rate -- High Steam Line Isolation	$\leq 12^{(9)}$

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq 12^{(3)}$
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
10. Loss-of-Offsite Power	
Motor-and Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60
11. Trip of All Main Feedwater Pumps	
Motor-Driven Auxiliary Feedwater Pumps	N.A.
12. Steam Line Differential Pressure--High Coincident with Main Steam Line Isolation Signal	
Isolate Auxiliary Feedwater to the Affected Steam Generator	$\leq$ 41
13. RWST Level--Low-Low	
a. Safety Injection Switchover to Containment Sump Coincident with Safety Injection	$\leq$ 32
b. Safety Injection Switchover to Containment Sump Coincident With Containment Spray	$\leq$ 103
14. Containment Radioactivity-High	
a. Normal Containment Purge Isolation	$\leq 3.5^{(7)}$
b. Preentry Containment Purge Isolation	$\leq 15^{(7)}$

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Applicable to main feedwater isolation valves and main feedwater bypass isolation valves only. Other valves that get a Feedwater Isolation signal (e.g., ammonia and hydrazine connections) are covered in Table 3.6-1.

(4) Diesel generator starting and sequence loading delay included. RHR pumps not included.

(5) Diesel generator starting and sequence loading delays not included. RHR pumps not included.

(6) Isolation of Normal Containment Purge. This value is not applicable to Pre-entry Containment Purge which is permitted to be operating only in MODES 5 or 6 as per Specification 3.6.1.7.

(7) Response time testing of radiation monitors is not required.

(8) Diesel generator starting delay not included, but sequencer loading delays are included.

(9) Applicable to main steam isolation valves and main steam bypass isolation valves only. Other valves that get a Main Steam Isolation signal (e.g., main steam drains to the condenser) are covered in Table 3.6-1.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure--High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
(2) Preentry Purge Detector	See Table 4.3-3, Item 1b2, for surveillance requirements.							
c) Airborne Particulate Radioactivity								
(1) RCS Leak Detection (normal purge)	See Table 4.3-3, Item 1C1, for surveillance requirements.							
(2) Preentry Purge Detector	See Table 4.3-3, Item 1C2, for surveillance requirements.							
5) Manual Phase A Isolation	See Item 3.a.1) above for Manual Phase A Isolation Surveillance Requirements.							
4. Main Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-- High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Main Steam Line Isolation (Continued)								
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Surveillance Requirements.							
e. Negative Steam Line Pressure Rate--High	S	R	M	N.A.	N.A.	N.A.	N.A.	3**, 4**
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level--High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Power Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Containment Spray Switchover to Containment Sump (Continued)								
b. RWST Level--Low-Low	See Item 7.b. above for RWST Level--Low-Low Surveillance Requirements.							
Coincident with Containment Spray	See Item 2. above for Containment Spray Surveillance Requirements.							
9. Loss-of-Offsite Power								
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	R	N.A.	M*	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	R.	N.A.	M*	N.A.	N.A.	N.A.	1, 2, 3, 4
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
Not P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) The Surveillance Requirements of Specification 4.9.9 apply during CORE ALTERATIONS or movement of irradiated fuel in containment.

(3) Except for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 which shall be tested at least once per 18 months and during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days.

* Setpoint verification not required.

\# During CORE ALTERATIONS or movement of irradiated fuel in containment.

** Trip Function automatically blocked above P-11 and may be blocked below P-11 when safety injection or low steamline pressure is not blocked.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment Radioactivity--					
a. Containment Ventilation Isolation Signal Area Monitors	2	3	1, 2, 3, 4, 6	#	27
b. Airborne Gaseous Radioactivity					
1) RCS Leakage Detection	1	1	1, 2, 3, 4	$\leq 1.0 \times 10^{-3}$ μCi/ml	26, 27
2) Pre-entry Purge	1	1	##	$\leq 2.0 \times 10^{-3}$ μCi/ml	30
c. Airborne Particulate Radioactivity					
1) RCS Leakage Detection	1	1	1, 2, 3, 4	$\leq 4.0 \times 10^{-8}$ μCi/ml	26, 27
2) Pre-entry Purge	1	1	##	$\leq 1.5 \times 10^{-8}$ μCi/ml	30
2. Spent Fuel Pool Area--Fuel Handling Building Emergency Exhaust Actuation					
a. Fuel Handling Building Operating Floor--South Network	1***	2	**	$\leq$ 100 mR/hr	28
b. Fuel Handling Building Operating Floor--North Network	1***	2	*	$\leq$ 100 mR/hr	28
3. Control Room Outside Air Intakes--					
a. Normal Outside Air Intake Isolation	1	2	All	$\leq 4.9 \times 10^{-6}$ μCi/ml	29

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
3. Control Room Outside Air Intakes-- (Continued)					
b. Emergency Outside Air Intake Isolation--South Intake	1	2	All	$\leq 4.9\times10^{-6}$ µCi/ml	29
c. Emergency Outside Air Intake Isolation--North Intake	1	2	All	$\leq 4.9\times10^{-6}$ µCi/ml	29

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TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

** With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

*** Each channel consists of 3 detectors with 1 of 3 logic. A channel is OPERABLE when 1 or more of the detectors are OPERABLE.

\# For MODES 1, 2, 3 and 4, the setpoint shall be less than or equal to three times detector background at RATED THERMAL POWER. During fuel movement the setpoint shall be less than or equal to 150 mR/hr.

\#\# Required OPERABLE whenever pre-entry purge system is to be used.

ACTION STATEMENTS

ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, suspend all operations involving movement of fuel within the storage pool or crane operations over the storage pool.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour initiate isolation of the respective air intake. With no outside air intakes available, maintain operation of the Control Room Emergency Filtration System in the Recirculation Mode of Operation.

ACTION 30 - With less than the minimum channels OPERABLE requirement, pre-entry purge operations shall be suspended and the containment pre-entry purge makeup and exhaust valves shall be maintained closed.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.a The Remote Shutdown System monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels required by Table 3.3-9, restore the inoperable channels to OPERABLE status within 60 days or submit a Special Report in accordance with Specification 6.9.2 within 14 additional days.

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit and control switch required by 3.3.3.5.b, shall be demonstrated OPERABLE at least once per 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant System Hot-Leg Temperature	ACP*	2	2
2. Reactor Coolant System Cold-Leg Temperature	ACP*	2	2
3. Pressurizer Pressure	ACP*	2	1-SSA Channel**
4. Pressurizer Level	ACP*	2	1-SSA Channel**
5. Steam Generator Pressure (Note 1)	ACP*	1/Steam Generator	1/Steam Generator
6. Steam Generator Water Level--Wide Range (Note 1)	ACP*	1/Steam Generator	1/Steam Generator
7. Residual Heat Removal Flow	ACP*	2	1 (Note 2)
8. Auxiliary Feedwater Flow (Note 1)	ACP*	1/Steam Generator	N.A. (Note 3)
9. Condensate Storage Tank Level	ACP*	2	1-SSA Channel**
10. Reactor Coolant System Pressure-Wide Range	ACP*	2	1-SSA Channel**
11. Wide-Range Flux Monitor (SR Indicator)	ACP*	1	1-SSA Channel**
12. Charging Header Flow	ACP*	1	1-SSA Channel**
13. a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP	ACP*	1	1-SSA Channel**
or			
b. Auxiliary Feedwater Turbine Speed	ACP*	1	1-SSA Channel**
14. Boric Acid Tank Level	ACP*	1	1-SSA Channel**

*ACP = Auxiliary Control Panel
**SSA = Safe Shutdown Analysis

Note 1 - Steam Generators A&B Only
Note 2 - RHR Train B Only
Note 3 - Steam Generator Water Level is used

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TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant System Hot-Leg Temperature	M	R
2. Reactor Coolant System Cold-Leg Temperature	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Pressure	M	R
6. Steam Generator Water Level--Wide Range	M	R
7. Residual Heat Removal Flow	M	R
8. Auxiliary Feedwater Flow	M	R
9. Condensate Storage Tank Level	M	R
10. Reactor Coolant System Pressure--Wide Range	M	R
11. Wide-Range Flux Monitor (SR Indicator)	M	Q
12. Charging Header Flow	M	R
13. a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP	M	R
b. Auxiliary Feedwater Turbine Speed	M	R
14. Boric Acid Tank Level	M	R

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INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Required Number of Channels shown in Table 3.3-10, except for the pressurizer safety valve position indicator or the sub-cooling margin monitor, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the pressurizer safety valve position indicator, or the sub-cooling margin monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

c. With the number of OPERABLE channels for the radiation monitors, the pressurizer safety valve position indicator*, or the sub-cooling margin monitor#, less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

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* The alternate method shall be a check of safety valve piping temperatures and evaluation to determine position.

The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

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INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

Specification 3/4 3.3.8 DELETED
Table 3.3-11 DELETED

INSTRUMENTATION

METAL IMPACT MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.9 The Metal Impact Monitoring System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Metal Impact Monitoring System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the Metal Impact Monitoring System shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST, except for verification of setpoint, at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/ Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately (1) suspend the release of radioactive liquid effluents monitored by the affected channel or (2) declare the channel inoperable and take ACTION as directed by b. below.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Exert best effort to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4, are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Lines		
1) Treated Laundry and Hot Shower Tanks Discharge Monitor	1	35
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor	1	35
3) Secondary Waste Sample Tank Discharge Monitor	1	35
b. Turbine Building Floor Drains Effluent Line	1	36
c. Outdoor Tank Area Drain Transfer Pump Monitor	1	37
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Normal Service Water System Return From Waste Processing Building to the Circulating Water System	1	39
b. Normal Service Water System Return From the Reactor Auxiliary Building to the Circulating Water System	1	39
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Lines		
1) Treated Laundry and Hot Shower Tanks Discharge	1	38
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge	1	38

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TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
3. Flow Rate Measurement Devices (Continued)		
3) Secondary Waste Sample Tank	1	38
b. Cooling Tower Blowdown	1	38

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluents releases via this pathway may continue provided the weekly Cooling Tower Blowdown weir surveillance is performed as required by Specification 4.11.1.1.1. Otherwise follow the ACTION specified in ACTION 37 above..

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST
1. Radioactiity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Lines				
1) Treated Laundry and Hot Shower Tanks Discharge Monitor	D	P	R(3)	Q(1)
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor	D	P	R(3)	Q(1)
3) Secondary Waste Sample Tank Discharge Monitor	D	P	R(3)	Q(1)
b. Turbine Building Floor Drains Effluent Line	D	M	R(3)	Q(1)
c. Outdoor Tank Area Drain Transfer Pump Monitor	D	M	R(3)	Q(1)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Normal Service Water System Return From the Waste Processing Building to the Circulating Water System	D	M	R(3)	Q(2)

TABLE 4.3-8 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release (Continued)				
b. Normal Service Water System Return From the Reactor Auxiliary Building to the Circulating Water System	D	M	R(3)	Q(2)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Lines				
1) Treated Laundry and Hot Shower Tanks Discharge	D(4)	N.A.	R	N.A.
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge	D(4)	N.A.	R	N.A.
3) Secondary Waste Sample Tank	D(4)	N.A.	R	N.A.
b. Cooling Tower Blowdown	D(4)	N.A.	R	N.A.

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (monitor loss of communications (alarm only), detector loss of counts (Alarm only) or monitor loss of power), or

c. Detector check source test failure (alarm only), or

d. Detector channel out of service (alarm only), or

e. Monitor loss of sample flow (alarm only).

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (monitor loss of communications (alarm only), detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out of service, or

e. Monitor loss of sample flow.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately (1) suspend the release of radioactive gaseous effluents monitored by the affected channel or (2) declare the channel inoperable and take ACTION as directed by b. below.

b. With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instrument to OPERABLE status within 30 days. If unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and a DIGITAL CHANNEL OPERATIONAL TEST or an ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS WASTE PROCESSING SYSTEM--HYDROGEN AND OXYGEN ANALYZERS			
a. Recombiner Outlet Hydrogen Monitor	1/recombiner	**	50
b. Recombiner Outlet Oxygen Monitor	1/recombiner	**	48
c. Compressor Discharge Oxygen Monitor	1	**	48
2. TURBINE BUILDING VENT STACK			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	49
c. Particulate Sampler	1	*	49
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46
3. PLANT VENT STACK			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	49
c. Particulate Sampler	1	*	49
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4.	WASTE PROCESSING BUILDING VENT STACK 5			
a.1	Noble Gas Activity Monitor (PIG)	1	*	45, 51
a.2	Noble Gas Activity Monitor (WRGM)	1	MODES 1, 2, 3	52
b.	Iodine Sampler	1	*	49
c.	Particulate Sampler	1	*	49
d.	Flow Rate Monitor	1	*	46
e.	Sampler Flow Rate Monitor	1	*	46
5.	WASTE PROCESSING BUILDING STACK 5A			
a.	Noble Gas Activity Monitor	1	*	47
b.	Iodine Sampler	1	*	49
c.	Particulate Sampler	1	*	49
d.	Flow Rate Monitor	1	*	46
e.	Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE TREATMENT operation

ACTION STATEMENTS

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the waste gas decay tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - With the number of channnels OPERABLE less than the Minimum Channels OPERABLE requirement, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 50 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement for both the PIG and WRGM, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 52 - Take the ACTION as required by Specification 3.3.3.6 ACTION c.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS WASTE PROCESSING SYSTEM-- HYDROGEN AND OXYGEN ANALYZERS					
a. Recombiner Outlet Hydrogen Monitor	D	N.A.	Q(4)	M#	**
b. Recombiner Outlet Oxygen Monitor	D	N.A.	Q(5)	M#	**
c. Compressor Discharge Oxygen Monitor	D	N.A.	Q(5)	M#	**
2. TURBINE BUILDING VENT STACK					
a. Noble Gas Activity	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
3. PLANT VENT STACK					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. WASTE PROCESSING BUILDING VENT STACK 5					
a.1 Noble Gas Activity Monitor (PIG)	D	M	R(3)	Q(1)	*
a.2 Noble Gas Activity Monitor (WRGM)	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
5. WASTE PROCESSING BUILDING VENT STACK 5A					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3-9 (Continued)

TABLE NOTATIONS

*At all times.

**During GASEOUS RADWASTE TREATMENT SYSTEM operation.

#ANALOG CHANNEL OPERATIONAL TEST

(1) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (monitor loss of communications - (alarm only), detector loss of counts (alarm only) or monitor loss of power), or

c. Detector check source test failure (alarm only), or

d. Detector channel out of service (alarm only), or

e. Monitor loss of sample flow (alarm only)

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (monitor loss of communications (alarm only), detector loss of counts, or monitor loss of power), or

c. Detector check source test failure, or

d. Detector channel out of service, or

e. Monitor loss of sample flow.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing hydrogen and nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing oxygen and nitrogen.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one throttle valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 31 days by direct observation of the movement of each of the following valves through at least one complete cycle from the running position:

1. Four high pressure turbine throttle valves,
2. Four high pressure turbine governor valves,
3. Four low pressure turbine reheat stop valves, and
4. Four low pressure turbine reheat intercept valves.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

c. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

*Not applicable in MODE 2 or 3 with all main steam isolation valves and bypass valves in the closed position and all other steam flow paths to the turbine isolated.

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REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant pumps in operation when the Reactor Trip System breakers are closed or with one reactor coolant pump in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, immediately open the Reactor Trip System breakers, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

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*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**See Special Test Exception 3.10.4.

FINAL DRAFT

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable as a result of causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve.

c. With two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With all three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

e. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b., c. or d. above, as appropriate, for the isolated PORV(s).

f. The provisions of Specification 3.0.4 are not applicable.

* MODE 4 when the temperature of all RCS cold legs is greater than 335°F.

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment Airborne Gaseous or Particulate Radioactivity Monitor at least once per 12 hours;

b. Monitoring the containment sump inventory and Flow Monitoring System at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 335°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of at least one vent valve and one block valve, powered from emergency buses, shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and
b. Pressurizer steam space

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuators of all the vent valves in the inoperable vent path and both block valves; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, due to causes other than the removal of power to both block valves pursuant to Action a, maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

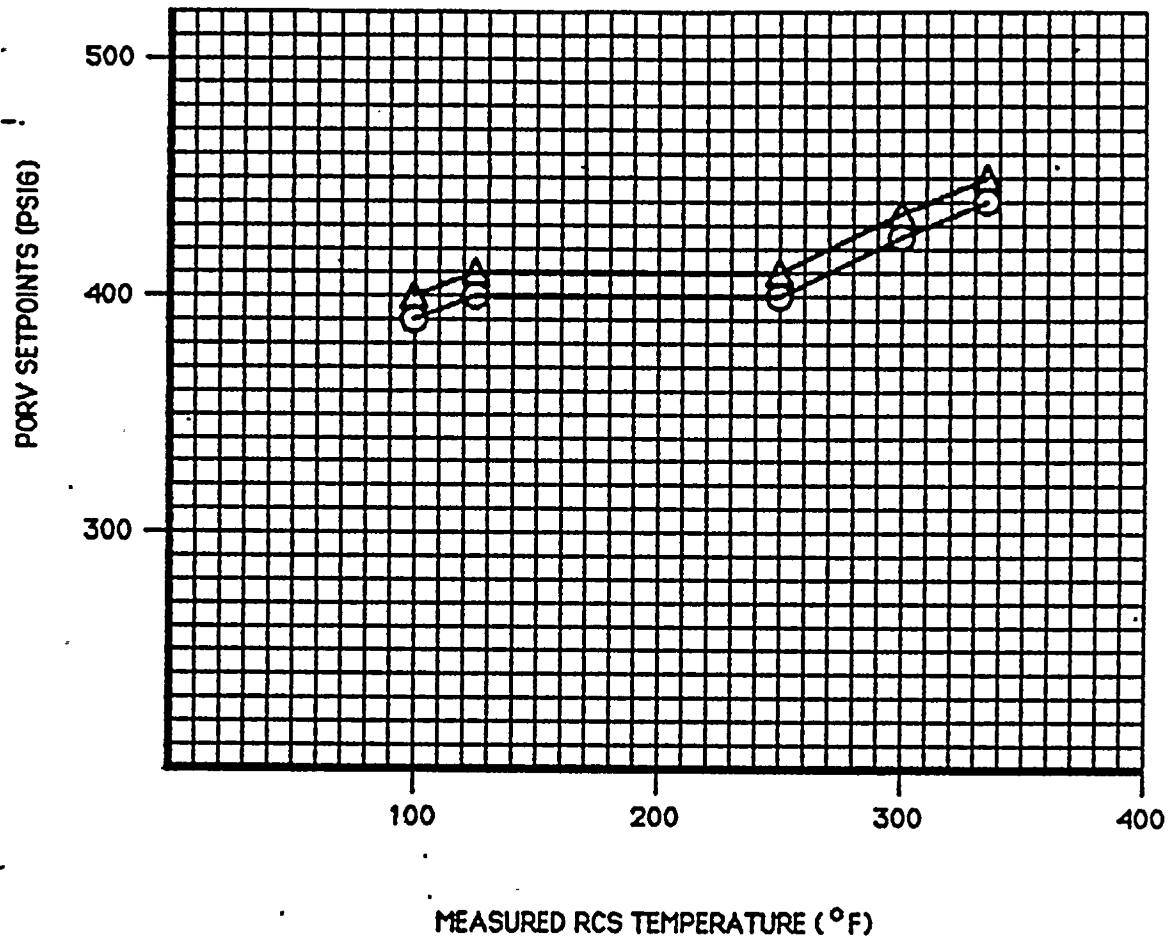
4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.



RCS TEMP °F	LOW PORV * PSIG ○	HIGH PORV * PSIG Δ
< 100	390	400
125	400	410
250	400	410
300	425	435
335	440	450

* VALUES BASED ON 4 EFPY REACTOR VESSEL DATA AND CONTAINS MARGINS OF −10°F AND +60 PSIG FOR POSSIBLE INSTRUMENT ERROR

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open with power supply circuit breaker open,

b. A contained borated water volume of between 66 and 96% indicated level,

c. A boron concentration of between 2000 and 2200 ppm, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*RCS pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on safety injection actuation test signal and on safety injection switchover to containment sump from an RWST Lo-Lo level test signal, and

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a) charging/safety injection pump,
b) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure when tested pursuant to Specification 4.0.5:

1. charging/safety injection pump (Refer to Specification 4.1.2.4)
2. RHR pump ≥ 100 psid at a flow rate of at least 3663 gpm.

g. By verifying that the locking mechanism is in place and locked for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2. At least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

HPSI SYSTEM EBASCO Valve No.	HPSI SYSTEM CP&L Valve No.
2SI-V440SA-1	1SI-5
2SI-V439SB-1	1SI-6
2SI-V438SA-1	1SI-7
2SI-V437SA-1	1SI-69
2SI-V436SB-1	1SI-70
2SI-V435SA-1	1SI-71
2SI-V434SA-1	1SI-101
2SI-V433SB-1	1SI-102
2SI-V432SA-1	1SI-103
2SI-V431SA-1	1SI-124
2SI-V430SB-1	1SI-125
2SI-V429SA-1	1SI-126

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For charging/safety injection pump lines, with a single pump running:

   a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 379 gpm, and

   b) The total pump flow rate is less than or equal to 685 gpm.

2. For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

FINAL DRAFT

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations*# not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 41 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

#Valves CP-B3, CP-B7 and CM-B5 may be verified at least once per 31 days by manual remote keylock switch position.

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SURVEILLANCE REQUIREMENTS (Continued)

period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the following equation:

      $|L_c - (L_{am} + L_o)| \leq 0.25\ L_a$, where L_{am} is the measured Type A test leakage and L_o is the superimposed leak;

   2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3. Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Containment purge makeup and exhaust isolation valves with resilient material seals,

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge makeup and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2;

g. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Except during entry to repair an inoperable inner door, for a cumumlative time not to exceed one hour per year.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours, to be within the limit:

Location

a. Elevation 290 ft
b. Elevation 240 ft
c. Elevation 230 ft

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1. Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined, during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2), by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports. Any abnormal degradation of the containment vessel structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge makeup and exhaust isolation valve shall be OPERABLE and:

a. Each 42-inch containment preentry purge makeup and exhaust isolation valve shall be closed and sealed closed, and

b. The 8-inch containment purge makeup and exhaust isolation valve(s) may be open for safety-related reasons only.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a 42-inch containment preentry purge makeup and/or exhaust isolation valve open or not sealed closed, close and/or seal close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 8-inch containment purge makeup and/or exhaust isolation valve(s) inoperable for any reason other than leakage integrity, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge makeup and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying that, on an indicated recirculation flow of at least 2150 gpm, each pump develops a discharge pressure of greater than or equal to 229 psig when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal and

   2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

   3. Verifying that, coincident with an indication of containment spray pump running, each automatic valve from the sump and RWST actuates to its appropriate position following an RWST Lo-Lo test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 2736 and 2912 gallons of between 28% and 30% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and
2. Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a containment spray or containment isolation phase A test signal as applicable; and

d. At least once per 5 years by verifying each eductor flow rate is between 19.5 and 20.5 gpm, using the RWST as the test source containing at least 436,000 gallons of water.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Four containment fan coolers (AH-1, AH-2, AH-3 and AH-4) shall be OPERABLE with one of two fans in each cooler capable of operation at low speed. Train SA consists of AH-2 and AH-3. Train SB consists of AH-1 and AH-4.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required trains of fan coolers to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one train of the above required containment fan coolers inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable train of containment fan coolers to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment fan coolers shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each fan train from the control room, and verifying that each fan train operates for at least 15 minutes, and

2. Verifying a cooling water flow rate of greater than or equal to 1425 gpm to each cooler.

b. At least once per 18 months by verifying that each fan train starts automatically on a safety injection test signal.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
78C	1SP-59 (SP-V1)	PRESSURIZER STEAM SAMPLE	<60	2
78C	1SP-60 (SP-V2)	PRESSURIZER STEAM SAMPLE	<60	2
78D	1SP-78 (SP-V113)	ACCUMULATOR SAMPLE	<60	2
78D	1SP-81 (SP-V114)	ACCUMULATOR SAMPLE	<60	2
78D	1SP-84 (SP-V115)	ACCUMULATOR SAMPLE	<60	2
78D	1SP-85 (SP-V116)	ACCUMULATOR SAMPLE	<60	2
80	1IA-819 (IA-V192)	INSTRUMENT AIR SUPPLY	<60	N/A
88	1SP-201 (SP-V406)	LIQUID SAMPLE RETURN FROM PASS SKID #1	5	3
88	1SP-200 (SP-V407)	LIQUID SAMPLE RETURN FROM PASS SKID #1	5	3
91	1SW-240 (SW-B89)	SERVICE WATER FROM NNS FAN COILS	<60	2
91	1SW-242 (SW-B90)	SERVICE WATER FROM NNS FAN COILS	<60	2
92	1SW-231 (SW-B88)	SERVICE WATER TO NNS FAN COILS	<60	2
105	1FP-347 (FP-B1)	FIRE WATER SPRINKLER SUPPLY	<60	2
108	1AF-155 (AF-V162)	AUX. F.W. TO S/G A (HYDRAZINE)	10	1,2,6
108	1AF-153 (AF-V163)	AUX. F.W. TO S/G A (AMMONIA)	10	1,2,6

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
109	1AF-159 (AF-V164)	AUX. F.W. TO S/G B (HYDRAZINE)	10	1,2,6
109	1AF-157 (AF-V165)	AUX. F.W. TO S/G B (AMMONIA)	10	1,2,6
110	1AF-163 (AF-V166)	AUX. F.W. TO S/G C (HYDRAZINE)	10	1,2,6
110	1AF-161 (AF-V167)	AUX. F.W. TO S/G C (AMMONIA)	10	1,2,6
73A	1SP-12 (SP-V300)	H_2 ANALYZER	<60	2,3
73A	1SP-915 (SP-V348)	H_2 ANALYZER	<60	2,3
73B	1SP-941 (SP-V301)	H_2 ANALYZER	<60	2,3
73B	1SP-917 (SP-V349)	H_2 ANALYZER	<60	2,3
83A	1SP-916 (SP-V448)	RADIATION MONITOR	<60	2
83A	1SP-16 (SP-V449)	RADIATION MONITOR	<60	2
83B	1SP-918 (SP-V450)	RADIATION MONITOR	<60	2
83B	1SP-939 (SP-V451)	RADIATION MONITOR	<60	2
86A	1SP-42 (SP-V308)	HYDROGEN ANALYZER	<60	2,3
86A	1SP-919 (SP-V314)	HYDROGEN ANALYZER	<60	2,3
86B	1SP-62 (SP-V309)	HYDROGEN ANALYZER	<60	2,3
86B	1SP-943 (SP-V315)	HYDROGEN ANALYZER	<60	2,3

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
2. PHASE B ISOLATION				
35	1CC-208 (CC-V170)	CCW TO RCP	10	N/A
36	1CC-297 (CC-V184)	CCW FROM RCP	10	N/A
36	1CC-299 (CC-V183)	CCW FROM RCP	10	N/A
39	1CC-249 (CC-V191)	CCW FROM RCP THERMAL BARRIERS	10	N/A
39	1CC-251 (CC-V190)	CCW FROM RCP THERMAL BARRIERS	10	N/A
3. SAFETY INJECTION ACTUATION				
8	1CS-238 (CS-V610)	CVCS NORMAL CHARGING	10	N/A
51	1BD-11 (BD-V11)	S/G A BLOWDOWN	<60	1,2,6
52	1BD-30 (BD-V15)	S/G B BLOWDOWN	<60	1,2,6
53	1BD-49 (BD-V19)	S/G C BLOWDOWN	<60	1,2,6
54	1SP-217 (SP-V120)	S/G A SAMPLE	<60	1,2,6
55	1SP-222 (SP-V121)	S/G B SAMPLE	<60	1,2,6
56	1SP-227 (SP-V122)	S/G C SAMPLE	<60	1,2,6
4. CONTAINMENT VENTILATION ISOLATION				
57	CP-9 (CP-B1)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5
57	CP-10 (CP-B3)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	2,5

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
57	CP-7 (CP-B4)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	2,5
57	CP-6 (CP-B2)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5
58	CP-4 (CP-B7)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	2,5
58	CP-5 (CP-B5)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5
58	CP-1 (CP-B8)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	2,5
58	CP-3 (CP-B6)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5
59	CB-2 (CB-B1)	CONTAINMENT VACUUM RELIEF	~~5~~	~~3~~
98	CB-6 (CB-B2)	CONTAINMENT VACUUM RELIEF	5	3
5. CONTAINMENT SPRAY ACTUATION				
23	1CT-50 (CT-V21)	CONTAINMENT SPRAY	N/A	3
24	1CT-88 (CT-V43)	CONTAINMENT SPRAY	N/A	3
6. MAIN STEAM LINE ISOLATION				
3	1MS-80 (MS-V1)	MSIV (S/G A)	5	1,4
3	1MS-81 (MS-F1)	MSIV BYPASS	10	1,2,3,6
3	1MS-231 (MS-V59)	MS DRAIN TO CONDENSER	<60	1,2,6
2	1MS-82 (MS-V2)	MSIV (S/G B)	5	1,4

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Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
2	1MS-83 (MS-F2)	MSIV BYPASS	10	1,2,3,6
2	1MS-266 (MS-V60)	MS DRAIN TO CONDENSER	<60	1,2,6
1	1MS-84 (MS-V3)	MSIV (S/G C)	5	1,4
1	1MS-85 (MS-F3)	MSIV BYPASS	10	1,2,3,6
1	1MS-301 (MS-V61)	MS DRAIN TO CONDENSER	<60	1,2,6
7. MAIN FEEDWATER LINE ISOLATION				
4	1FW-159 (FW-V26)	FEEDWATER LOOP A	5	1,2,6
4	1FW-307 (FW-V123)	FEEDWATER LOOP A BYPASS VALVE	10	1,2,6
4	1FW-165 (FW-V89)	FEEDWATER LOOP A (HYDRAZINE)	<60	1,2,6
4	1FW-163 (FW-V90)	FEEDWATER LOOP A (AMMONIA)	<60	1,2,6
5	1FW-277 (FW-V27)	FEEDWATER LOOP B	5	1,2,6
5	1FW-319 (FW-V124)	FEEDWATER LOOP B BYPASS VALVE	10	1,2,6
5	1FW-279 (FW-V91)	FEEDWATER LOOP B (AMMONIA)	<60	1,2,6
5	1FW-281 (FW-V92)	FEEDWATER LOOP B (HYDRAZINE)	<60	1,2,6
6	1FW-217 (FW-V28)	FEEDWATER LOOP C	5	1,2,6
6	1FW-313 (FW-V125)	FEEDWATER LOOP C BYPASS VALVE	10	1,2,6

FINAL DRAFT

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
6	1FW-223 (FW-V93)	FEEDWATER LOOP C (AMMONIA)	<60	1,2,6
6	1FW-221 (FW-V94)	FEEDWATER LOOP C (HYDRAZINE)	<60	1,2,6
108	1AF-64 (AF-V156)	AUXILIARY FEEDWATER A PREHEATER BYPASS	10	1,2,6
109	1AF-102 (AF-V157)	AUXILIARY FEEDWATER B PREHEATER BYPASS	10	1,2,6
110	1AF-81 (AF-V158)	AUXILIARY FEEDWATER C PREHEATER BYPASS	10	1,2,6
8. AUXILIARY FEEDWATER ISOLATION				
108	1AF-55 (AF-V10)	AUXILIARY FEEDWATER TO S/G A	24	1,6
108	1AF-137 (AF-V116)	AUXILIARY FEEDWATER TO S/G A	24	1,6
109	1AF-93 (AF-V19)	AUXILIARY FEEDWATER TO S/G B	24	1,6
109	1AF-143 (AF-V117)	AUXILIARY FEEDWATER TO S/G B	24	1,6
110	1AF-74 (AF-V23)	AUXILIARY FEEDWATER TO S/G C	24	1,6
110	1AF-149 (AF-V118)	AUXILIARY FEEDWATER TO S/G C	24	1,6
9. REMOTE MANUAL VALVES				
3	1MS-58 (MS-P18)	S/G PORV (S/G A)	N/A	1,2,3,6
2	1MS-60 (MS-P19)	S/G PORV (S/G B)	N/A	1,2,3,6
1	1MS-62 (MS-P20)	S/G PORV (S/G C)	N/A	1,2,3,6

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
9	1CS-341 (CS-V522)	CVCS SEAL WATER TO RCP A	N/A	N/A
10	1CS-382 (CS-V523)	CVCS SEAL WATER TO RCP B	N/A	N/A
11	1CS-423 (CS-V524)	CVCS SEAL WATER TO RCP C	N/A	N/A
13	1SI-340 (SI-V579)	SI-LOW HEAD TO COLD LEGS	N/A	1
14	1SI-341 (SI-V578)	SI-LOW HEAD TO COLD LEGS	N/A	1
15	1RH-2 (RH-V503)	RHR PUMP SUCTION (TRAIN A)	N/A	1,3
16	1RH-40 (RH-V501)	RHR PUMP SUCTION (TRAIN B)	N/A	1,3
18	1SI-359 (SI-V587)	SI LOW HEAD TO HOT LEG	N/A	3
20	1SI-107 (SI-V500)	SI HIGH HEAD TO HOT LEG	N/A	3
21	1SI-86 (SI-V501)	SI HIGH HEAD TO HOT LEG	N/A	3
22	1SI-52 (SI-V502)	SI HIGH HEAD TO COLD LEG	N/A	3
25	1SW-92 (SW-B46)	SERVICE WATER TO FAN COOLER AH-3	N/A	1,6
26	1SW-91 (SW-B45)	SERVICE WATER TO FAN COOLER AH-2	N/A	1,6
27	1SW-225 (SW-B52)	SERVICE WATER TO FAN COOLER AH-1	N/A	1,6
28	1SW-227 (SW-B51)	SERVICE WATER TO FAN COOLER AH-4	N/A	1,6
29	1SW-97 (SW-B47)	SERVICE WATER FROM FAN COOLER AH-3	N/A	1,6

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
30	1SW-109 (SW-B49)	SERVICE WATER FROM FAN COOLER AH-2	N/A	1,6
31	1SW-98 (SW-B48)	SERVICE WATER FROM FAN COOLER AH-1	N/A	1,6
32	1SW-110 (SW-B50)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1,6
17	1SI-3 (SI-V505)	SI TO HIGH HEAD COLD LEG	N/A	3
17	1SI-4 (SI-V506)	SI TO HIGH HEAD COLD LEG	N/A	3
2	1MS-70 (MS-V8)	MAIN STEAM B TO AUXILIARY F.W. TURBINE	N/A	1,3,6
1	1MS-72 (MS-V9)	MAIN STEAM C TO AUXILIARY F.W. TURBINE	N/A	1,3,6
63	CM-2 (CM-B5)	H_2 PURGE EXHAUST	N/A	3
10. MANUAL VALVES				
17	1SI-43 (SI-V30)	SI-HIGH HEAD TO COLD LEGS	N/A	1,3
34	1LT-6 (LT-V2)	ILRT ROTOMETER (LOCKED CLOSED)	N/A	2,3
41	1SA-80 (SA-V14)	SERVICE AIR (LOCKED CLOSED)	N/A	2,3
42	1ED-119 (WL-D651)	RCDT PUMP DISCH BYPASS (LOCKED CLOSED)	N/A	2,3
44	1SF-145 (SF-D164)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
44	1SF-144 (SF-D165)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
45	1SF-118 (SF-D25)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3

FINAL DRAFT

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
45	1SF-119 (SF-D26)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	2,3
61	CM-5 (CM-B6)	H_2 PURGE MAKEUP (LOCKED CLOSED)	N/A	3
79	1FP-355 (FP-V44)	FIRE WATER STANDPIPE SUPPLY	N/A	2,3
62	1LT-10 (LT-V4)	ILRT (LOCKED CLOSED)	N/A	2,3
63	CM-4 (CM-B4)	H_2 PURGE EXHAUST (LOCKED CLOSED)	N/A	3
90	1DW-63 (DW-V120)	DEMIN WATER SUPPLY (LOCKED CLOSED)	N/A	2,3
96	1LT-4 (LT-V1)	ILRT (LOCKED CLOSED)	N/A	2,3
108	1AF-174 (AF-V189)	WET LAY-UP TO STM GEN A AF HEADER	N/A	1,2,6
109	1AF-173 (AF-V190)	WET LAY-UP TO STM GEN B AF HEADER	N/A	1,2,6
110	1AF-175 (AF-V191)	WET LAY-UP TO STM GEN C AF HEADER	N/A	1,2,6
11. CHECK VALVES				
8	1CS-477 (CS-V515)	CVCS NORMAL CHARGING	N/A	N/A
12	1CS-471 (CS-V67)	CVCS SEAL WATER RETURN & EXCESS LETDOWN	N/A	N/A
23	1CT-53 (CT-V27)	CONTAINMENT SPRAY TRAIN A	N/A	N/A
24	1CT-91 (CT-V51)	CONTAINMENT SPRAY TRAIN B	N/A	N/A
35	1CC-211 (CC-V171)	CCW TO RCP	N/A	N/A

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
36	1CC-298 (CC-V51)	CCW FROM RCP	N/A	N/A
39	1CC-250 (CC-V50)	CCW FROM RCP THERMAL BARRIER	N/A	N/A
40	1RC-164 (RC-V525)	DEMIN WATER TO PRT	N/A	N/A
41	1SA-82 (SA-V15)	SERVICE AIR	N/A	N/A
59	CB-3 (CB-V1)	CONTAINMENT VACUUM RELIEF	N/A	N/A
61	CM-7 (CM-V1)	H_2 PURGE MAKEUP	N/A	N/A
76A	1SI-182 (SI-V150)	ACCUMULATORY FILL FROM RWST	N/A	N/A
77A	1SI-290 (SI-V188)	N_2 TO ACCUMULATORS	N/A	N/A
79	1FP-357 (FP-V48)	FIRE WATER STANDPIPE SUPPLY	N/A	N/A
80	1AI-220 (AI-V33)	INSTRUMENT AIR SUPPLY	N/A	N/A
90	1DW-65 (DW-V121)	DEMIN WATER SUPPLY	N/A	N/A
92	1SW-233 (SW-V142)	SERVICE WATER TO NNS FAN COILS	N/A	N/A
94A	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
94B	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
94C	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
95A	(B)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1

Table 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES
98	CB-7	CONTAINMENT VACUUM RELIEF	N/A	N/A
95B	(B) (CB-V2)	EXCESS FLOW CHECK VALVE FOR CTMT VACUUM RELIEF SENSING	N/A	1
105	1FP-349 (FP-V46)	FIRE WATER SPRINKLER SUPPLY	N/A	N/A
12. RELIEF VALVES				
7	1CS-10 (CS-R500)	CVCS NORMAL LETDOWN	N/A	N/A
15	1RH-7 (RH-R501)	RHR SUCTION FROM HOT LEG	N/A	1
16	1RH-45 (RH-R500)	RHR SUCTION FROM HOT LEG	N/A	1
29	1SW-95 (SW-R1)	SERVICE WATER FROM FAN COOLER AH-3	N/A	1
30	1SW-107 (SW-R3)	SERVICE WATER FROM FAN COOLER AH-2	N/A	1
31	1SW-96 (SW-R2)	SERVICE WATER FROM FAN COOLER AH-1	N/A	1
32	1SW-108 (SW-R4)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1

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TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER			LIFT SETTING (± 1%)*	ORIFICE SIZE (IN.^{2})
STEAM GENERATOR				
A	B	C		
1MS-43	1MS-44	1MS-45	1170 psig	16.0
1MS-46	1MS-47	1MS-48	1185 psig	16.0
1MS-49	1MS-50	1MS-51	1200 psig	16.0
1MS-52	1MS-53	1MS-54	1215 psig	16.0
1MS-55	1MS-56	1MS-57	1230 psig	16.0

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

FINAL DRAFT

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1590 psig at a recirculation flow of greater than or equal to 50 gpm.

2. Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1510 psig on a recirculation flow of greater than or equal to 90 gpm when the secondary steam supply pressure is greater than 210 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

FINAL DRAFT

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1. Each automatic valve servicing safety-related equipment or isolating non-safety-related components actuates to its correct position on a Safety Injection test signal, and

2. Each Component Cooling Water System pump required to be OPERABLE starts automatically on a Safety Injection test signal.

3. Each automatic valve serving the gross failed fuel detector actuates to its correct position on a Low Surge Tank Level test signal.

*The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

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3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent emergency service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one emergency service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two emergency service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1. Each automatic valve servicing safety-related equipment or isolating non-safety portions of the system actuates to its correct position on a Safety Injection test signal, and

2. Each emergency service water pump and each emergency service water booster pump starts automatically on a Safety Injection test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum auxiliary reservoir water level at or above elevation 250 feet Mean Sea Level, USGS datum, and a minimum main reservoir water level at or above 205.7 feet mean sea level, USGS datum, and

b. A water temperature as measured at the respective intake structure of less than or equal to 95°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the water temperature and water level to be within their limits.

PLANT SYSTEMS

CONTROL ROOM EMERGENCY FILTRATION SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

Revisions 2, March 1978, and the system flow rate is 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980; and

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.1 inches water gauge while operating the system at a flow rate of 4000 cfm ± 10%;

2. Verifying that, on either a Safety Injection or a High Radiation test signal, the system automatically switches into an isolation with recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 315 cfm relative to adjacent areas during system operation;

4. Verifying that the heaters dissipate 14 ± 1.4 kW when tested in accordance with ANSI N510-1980; and

5. Verifying that, on a High Chlorine test signal, the system automatically isolates the control room within 15 seconds and initiates a recirculation flow through the HEPA filters and charcoal adsorber banks.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

PLANT SYSTEMS

3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent RAB Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one RAB Emergency Exhaust System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each RAB Emergency Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6800 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980;

   2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,

PLANT SYSTEMS

REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is less than 4.1 inches water gauge while operating the unit at a flow rate of 6800 cfm ± 10%,

2. Verifying that the system starts on a Safety Injection test signal,

3. Verifying that the system maintains the areas served by the exhaust system at a negative pressure of greater than or equal to 1/8 inch water gauge relative to the outside atmosphere,

4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the unit at a flow rate of 6800 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6800 cfm ± 10%.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

SURVEILLANCE REQUIREMENTS (Continued)

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*,**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

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*The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

**The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs that might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

FINAL DRAFT

3/4.7.10 FIRE SUPPRESSION - DELETED
3/4.7.11 FIRE RATED ASSEMBLIES - DELETED
TABLES 3.7-3, 3.7-4, 3.7-5 - DELETED

FINAL DRAFT

PLANT SYSTEMS

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

FINAL DRAFT

TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	MAXIMUM TEMPERATURE LIMIT (°F)
REACTOR AUXILIARY BUILDING	
1. Control Room Envelope, (El 305')	85
2. Process I&C, Room (El 305')	85
3. Rod Control Cabinets Area (El 305')	104
4. A&B Battery Rooms (El 286')	85
5. A&B Switchgear Rooms (El 286')	90
6. Main Steam, Feedwater Pipe Tunnel (El 286' & 261')	116
7. SA&SB Electrical Penetration Areas (El 261' & 286')	104
8. Area with MCC 1A35SA and 1B35SB (El 261')	104
9. HVAC Chillers, Auxiliary FW Piping & Valve Area (El 261')	104
10. CCW Pumps, CCW Hx, Auxiliary FW Pumps Area (El 236')	104
11. 1A-SA, 1B-SB, and 1C-SAB Charging Pump Rooms (El 236')	104
12. Service Water Booster Pump 1B-SB (El 236')	104
13. Mechanical and Electrical Penetration Areas (El 236')	104
14. Containment Spray Additive Tank, and H&V Equipment Area (El 216')	104
15. Trains A&B Containment Spray Pump, RHR Pump, H&V Equipment Areas (El 190')	104
FUEL HANDLING BUILDING	
16. Trains A&B Emergency Exhaust System Areas (El 261')	104
17. Fuel Pool Cooling Pump and Heat Exchanger Area (El 236')	104
WASTE PROCESSING BUILDING	
18. H&V Equipment Room (El 236')	104
MISCELLANEOUS	
19. Tank Area (El 236')	122
20. Diesel Fuel Oil Storage Building (El 242')	122
21. Emergency Service Water Electrical Equipment Room	104
22. Emergency Service Water Pump Room	122
23. DELETED	
24. 1A-SA & 1B-SB H&V Equipment Rooms (El 292')	122
25. 1A-SA & 1B-SB H&V Equipment Rooms (El 280')	118
26. 1A-SA & 1B-SB Electrical Rooms (El 261')	116
27. 1A-SA & 1B-SB Diesel Generator Rooms (El 261')	120

FINAL DRAFT

PLANT SYSTEMS

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 At least two independent Essential Services Chilled Water System loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.13 The Essential Services Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by Specification 4.0.5, and

b. At least once per 18 months by demonstrating that:

1. Non-essential portions of the system are automatically isolated upon receipt of a Safety Injection actuation signal, and

2. The system starts automatically on a Safety Injection actuation signal.

FINAL DRAFT

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 2670 gallons of fuel, which is equivalent to 85% indicated level,

2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

3. A separate fuel oil transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either emergency diesel generator (EDG) has not been successfully tested within the 24 hours preceding entry into this ACTION, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 separately for each such EDG within 24 hours. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 within 24 hours*#; restore the

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4 and a.6, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued):

diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See also ACTION d. below.

c. With one offsite circuit of 3.8.1.1.a and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 within 8 hours*#; restore one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. See also ACTION d. below. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Specification 3.8.1.1 ACTION a or b, as appropriate with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this ACTION for an OPERABLE diesel or a restored to OPERABLE diesel satisfies the EDG test requirement of ACTION a or b.

d. With one diesel generator inoperable, in addition to ACTION b and c above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE. If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, except as provided for in ACTION d.2 below.

2. If in MODES 1, 2, or 3 and the result of the inoperable diesel generator is that three auxiliary feedwater pumps are inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4 and a.6, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

e. With two of the required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by sequentially performing Surveillance Requirement 4.8.1.1.2.a.4 and a.6 on both diesels within 8 hours#, unless the diesel generators are already operating; restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION a. with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test(s) of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this ACTION for the OPERABLE diesels satisfies the EDG test requirement of ACTION a.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow ACTION b. with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and a.6 performed under this ACTION for a restored-to-OPERABLE diesel satisfies the EDG test requirement of ACTION b.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignment and power availability, and

b. Demonstrated OPERABLE at least once per 18 months by manually transferring the onsite Class 1E power supply from the unit auxiliary transformer to the startup auxiliary transformer.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4 and a.6, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the main fuel oil storage tank,

3. Verifying the fuel oil transfer pump can be started and transfers fuel from the storage system to the day tank,

4. Verifying the diesel generator can start** and accelerate to synchronous speed (450 rpm) with generator voltage and frequency 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds. Subsequently, verifying the generator is synchronized, gradually loaded** to an indicated 6200-6400 kW*** and operates for at least 60 minutes,

5. Verifying the pressure in at least one air start receiver to be greater than or equal to 190 psig, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

b. Check for and remove accumulated water:

1. From the day tank, at least once per 31 days and after each operation of the diesel where the period of operation was greater than 1 hour, and

2. From the main fuel oil storage tank, at least once per 31 days.

c. By sampling new fuel oil in accordance with ASTM-D4057-81 prior to addition to storage tanks and:

1. By verifying, in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks, that the sample has:

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 26 degrees but less than or equal to 38 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if the gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2. By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tank, in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A.

e. At least once per 184 days, on a STAGGERED TEST BASIS, the diesel generators shall be started** and accelerated to at least 450 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds after the start signal.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 6200-6400***kW in less than or equal to 60 seconds, and operate for at least 60 minutes. The diesel generator shall be started for this test by using one of the following signals on a rotating basis:

1. Simulated loss of offsite power by itself, and

2. A Safety Injection test signal by itself.

This test, if it is performed so that it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

f. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with the TDI Owners Group recommendations for this class of standby service.

2. Verifying that, on rejection of a load of greater than or equal to 1078 kW, the voltage and frequency are maintained with 6900 ± 690 volts and 60 ± 6.75 Hz, with frequency stabilizing to 60 ± 1.2 Hz within 10 seconds without any safety-related load tripping out or operating in a degraded condition.

3. Verifying that the load sequencing timer is OPERABLE with the interval between each load block within 10% of its design interval.

4. Simulating a loss of offsite power by itself, and:

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected shutdown loads through the load sequencer, and operating for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz.

5. Verifying that on a safety injection test signal (without loss of power) the diesel generator starts** on the auto-start signal and operates on standby for greater than or equal to 5 minutes.

6. Simulating a loss of offsite power in conjunction with a safety injection test signal, and

a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

b) Verifying the diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected emergency (accident) loads through the sequencing timers, and operating for greater than or equal to 5 minutes and maintaining the steady-state voltage and frequency at 6900 ± 690 volts and 60 ± 1.2 Hz.

c) Verifying that all diesel generator trips, except engine overspeed, loss of generator potential transformer circuit, generator differential, and emergency bus differential are automatically bypassed upon loss of offsite power signal in conjunction with a safety injection signal.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

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SURVEILLANCE REQUIREMENTS (Continued)

7. Verifying the diesel generator operates** for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 6800-7000 kW*** and, during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6200-6400 kW***. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4:8.1.1.2.f.6 b).#

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 6500 kW;

9. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Proceed through its shutdown sequence.

10. Verifying that the following diesel generator lockout features prevent diesel generator operation:

    a) Engine overspeed
    b) Generator differential
    c) Emergency bus differential
    d) Emergency Stop
    e) Operational and maintenance switch in the maintenance mode
    f) Loss of generator potential transformer circuit

11. Verifying the generator capability to reject a load of between 6200 and 6400 kW without tripping. The generator voltage shall not exceed 7590 volts during and following the load rejection;

12. Verifying that, with the diesel generator operating in a test mode and connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

#If Specification 4.8.1.1.2f.6 b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 6200-6400 kW for 1 hour or until operating temperature has stabilized.

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SURVEILLANCE REQUIREMENTS (Continued)

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 450 rpm in less than or equal to 10 seconds.

h. At least once per 10 years by:

1) Draining each main fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or other appropriate cleaning solution, and

2) Performing a pressure test, of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code, at a test pressure equal to 110% of the system design pressure.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day tank containing a minimum volume of 2670 gallons of fuel, which is equivalent to 85% indicated level,

2. A separate main fuel oil storage tank containing a minimum volume of 100,000 gallons of fuel, and

3. A fuel oil transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over irradiated fuel and within 8 hours, depressurize and vent the Reactor Coolant System through a vent of greater than or equal to 2.9 square inches. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1 and 4.8.1.1.2.

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ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the buses.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) given in Table 3.8-1 inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be demonstrated OPERABLE*:

a. At least once per 18 months:

1. By verifying that the 6900-volt circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

*Fuses are listed for Table completeness only; there are no surveillance requirements.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.4.1 (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
1	MOV-2CT-V6SA-1 (Motor) (Isolation Valve)	15 A Breaker	15 A Breaker
2	MOV-2CT-V6SA-1 (Valve Limit Switch Control)	8 A Fuse	15 A Breaker
3	MOV-2CT-V7SB-1 (Motor) (Isolation Valve)	15 A Breaker	15 A Breaker
4	MOV-2CT-V7SB-1 (Valve Limit Switch Control)	8 A Fuse	15 A Breaker
5	MOV-2SI-V571SA-1 (Motor) (Isolation Valve)	30 A Breaker	30 A Breaker
6	MOV-2SI-V571SA-1 (Valve Limit Switch-IND & ANN)	8 A Fuse	15 A Breaker
7	MOV-2SI-V570SB-1 (Motor) (Isolation Valve)	30 A Breaker	30 A Breaker
8	MOV-2SI-V570SB-1 (Valve Limit Switch-IND & ANN)	8 A Fuse	15 A Breaker
9	Containment Fan Cooler AH-37 (1A-NNS)	1600 A Switch Gear Breaker with 200 A sensor	400 A Fuse
10	Containment Fan Cooler AH-38 (1A-NNS)	1600 A Switch Gear Breaker with 200 A sensor	400 A Fuse
11	Containment Fan Cooler AH-39 (1A-NNS)	1600 A Switch Gear Breaker with 200 A sensor	400 A Fuse
12	Rod Control Drive Mech Fan E-80 (1A-NNS)	100 A Breaker	100 A Breaker
13	Rod Control Drive Mech Fan E-81 (1A-NNS)	100 A Breaker	100 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
29	Reactor Coolant Pump (1A-SN)	Relay Trips Feeder Breaker	Relay Trips Upstream Breaker
30	Lighting Panel (LP-105)	150 A Breaker at 120 v. A.C.	70 A Breaker at 480 v. A.C.
31	Lighting Panel (LP-106)	100 A Breaker at 120 v. A.C.	50 A Breaker at 480 v. A.C.
32	Lighting Panel (LP-101)	125 A Breaker at 120 v. A.C.	60 A Breaker at 480 v. A.C.
33	Lighting Panel (LP-102)	125 A Breaker at 120 v. A.C.	60 A Breaker at 480 v. A.C.
34	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
35	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
36	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
37	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
38	Elevator Disc Switch	100 A Breaker	100 A Breaker
39	Power Receptacles 1-2 & 1-6	60 A Breaker	60 A Breaker
40	Power Receptacles 1-9 & 1-13	60 A Breaker	60 A Breaker
41	Power Receptacles 1-10 & 1-14	60 A Breaker	60 A Breaker
42	Reactor Coolant Pump 1A-SN Oil BRG Lift Pump	30 A Breaker	30 A Breaker
43	Disk Switch for 5-Ton Monorail	50 A Breaker	50 A Breaker
44	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
45	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
46	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
47	Pressurizer Heater Back-Up (Group A)	90 A Breaker	100 A Fuse
48	Power Receptacles 1-1 & 1-5	60 A Breaker	60 A Breaker
49	Power Receptacles 1-17 & 1-74	60 A Breaker	60 A Breaker
50	Power Receptacles 1-18 & 1-75	60 A Breaker	60 A Breaker
51	Rod Position Indication Distribution Panel	50 A Breaker	100 A Breaker
52	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
53	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
54	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
55	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
56	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
57	Reactor Coolant Drain Tank Pump-1A	50 A Breaker	50 A Breaker
58	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
59	Pressurizer Heater Control Group C	90 A Breaker	100 A Fuse
60	Power Receptacles 1-76	60 A Breaker	60 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
61	Containment Circular Bridge Crane	225 A Breaker	225 A Breaker
62	IRVH Cable Bridge Hoist	15 A Breaker	15 A Breaker
63	AOV-1RC-P525SN-1	6 A Fuse	20 A Breaker
64	MOV-2SI-V537SA-1 (8808A) Pos. SW. ANN	3 A Fuse	15 A Breaker
65	Integrated Head Cooling Fan E-80 (1A-NNS)	20 A Breaker	20 A Breaker
66	Integrated Head Cooling Fan E-81 (1A-NNS)	20 A Breaker	20 A Breaker
67	AOV-2BD-F6SN-1 (PCV-8400A)	6 A Fuse	15 A Breaker
68	Damper (CV-D9-1)	6 A Fuse	15 A Breaker
69	Damper (CV-D13-1)	6 A Fuse	15 A Breaker
70	Con. Rod Drive Mech. Fan E-80 (1A-NNS)	6 A Fuse	15 A Breaker
71	Con. Rod Drive Mech. Fan E-81 (1A-NNS)	6 A Fuse	15 A Breaker
72	Reactor Coolant Pump (1A-SN) Space Heater	15 A Breaker	30 A Breaker
73	Inst. Rack C1-R1	20 A Breaker	20 A Breaker
74	AH-37 (1A-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
75	AH-38 (1A-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
76	AH-39 (1A-NNS) Motor Space Heater	15 A Breaker	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
77	Elevator Equipment Room Fan (E-3) (1X-NNS)	20 A Breaker	20 A Breaker
78	SV-7SP-V334-1 (Rad. Mon. Sampling Valves)	15 A Breaker	15 A Breaker
79	SV-7SP-V318-1	15 A Breaker	15 A Breaker
80	SV-7SP-V320-1	15 A Breaker	15 A Breaker
81	Containment Atmo. Rad. Mon. Valve (7SP-V322-1)	15 A Breaker	15 A Breaker
82	Containment Atmo. Rad. Mon. Valve (7SP-V324-1)	15 A Breaker	15 A Breaker
83	Containment Atmo. Rad. Mon. Valve (7SP-V326-1)	15 A Breaker	15 A Breaker
84	Containment Atmo. Rad. Mon. Valve (7SP-V328-1)	15 A Breaker	15 A Breaker
85	Containment Atmo. Rad. Mon. Valve (7SP-330-1)	15 A Breaker	15 A Breaker
86	Containment Atmo. Rad. Mon. Valve (7SP-332-1)	15 A Breaker	15 A Breaker
87	AOV-2RC-D528SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
88	MOV-2CS-V516SA-1 (8112) (Limit Switch)	8 A Fuse	15 A Breaker
89	AOV-2CS-V511SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
90	AOV-2CS-V512SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
91	AOV-2CS-V513SA-1 (Limit Switch)	8 A Fuse	15 A Breaker
92	MOV-2SI-V537SA-1 (8808A) (Limit Switch)	8 A Fuse	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
122	Cont. Fan Cooler AH-2 (1A-SA) Space Heater	15 A Breaker	15 A Breaker
123	Cont. Fan Cooler AH-2 (1B-SA) Space Heater	15 A Breaker	15 A Breaker
124	Cont. Fan Cooler AH-3 (1A-SA) Space Heater	15 A Breaker	15 A Breaker
125	Cont. Fan Cooler AH-3 (1B-SA) Space Heater	15 A Breaker	15 A Breaker
126	Primary Shield Cooling Fan S-2 (1A-SA) Heater	15 A Breaker	15 A Breaker
127	Hydrogen Recombiner	125 A Breaker	125 A Breaker
128	Reactor Support Cooling Fan S-4 (1A-SA)	100 A Breaker	100 A Breaker
129	MOV-1RH-V501SA-1 (8701B) (Isolation Valve)	15 A Breaker	15 A Breaker
130	MOV-2SI-V537SA-1 (8808A) (Accumulator "A" Discharge Valve)	40 A Breaker	40 A Breaker
131	MOV-2SI-V535SA-1 (8808C) (Accumulator "C" Discharge Valve)	40 A Breaker	40 A Breaker
132	MOV-2CS-V516SA-1 (8112) (RCP Seal Water Return Isolation Valve)	15 A Breaker	15 A Breaker
133	MOV-1RH-V503SA-1 (8701A) (RHRS Inlet Isolation Valve)	15 A Breaker	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
168	Valve 2SP-V86SB-1	6 A Fuse	15 A Breaker
169	Valve 2SP-V81SB-1	6 A Fuse	15 A Breaker
170	Valve 2SP-V1SB-1	6 A Fuse	15 A Breaker
171	Valve 2SP-V11SB-1	6 A Fuse	15 A Breaker
172	Valve 2SP-V111SB-1	6 A Fuse	15 A Breaker
173	Valve 2SP-V113SB-1	6 A Fuse	15 A Breaker
174	Valve 2SP-V114SB-1	6 A Fuse	15 A Breaker
175	Valve 2SP-V115SB-1	6 A Fuse	15 A Breaker
176	Cont. Fan Cooler AH-1 Damper CV-D1SB-1 Motor	6 A Fuse	20 A Breaker
177	Cont. Fan Cooler AH-1 Damper CV-D1SB-1 Motor	6 A Fuse	20 A Breaker
178	Cont. Fan Cooler AH-1 Damper CV-D1SB-1 Pos. Sw.	6 A Fuse	20 A Breaker
179	Cont. Fan Cooler AH-4 Damper CV-D7SB-1 Motor	6 A Fuse	20 A Breaker
180	Cont. Fan Cooler AH-4 Damper CV-D7SB-1 Motor	6 A Fuse	20 A Breaker
181	Cont. Fan Cooler AH-4 Damper CV-D7SB-1 Pos. Sw.	6 A Fuse	20 A Breaker
182	RA-1CR-356 1B-SB	0.6 A Fuse	15 A Breaker
183	RA-1CR-356 1D-SB	0.6 A Fuse	15 A Breaker
184	Valve 2SP-V90SB-1	8 A Fuse	15 A Breaker
185	Valve 2SP-V91SB-1	8 A Fuse	15 A Breaker
186	Valve 2SP-V85SB-1	8 A Fuse	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
220	Integrated Head Cooling Fan E-80 (1B-NSS)	6 A Fuse	15 A Breaker
221	Integrated Head Cooling Fan E-81 (1B-NSS)	6 A Fuse	15 A Breaker
222	Damper CV-D10-1	6 A Fuse	15 A Breaker
223	RCP-IB-SN Space Heater	15 A Breaker	30 A Breaker
224	Integrated Head Cooling Fan E-80 (1B-NNS)	20 A Breaker	20 A Breaker
225	Integrated Head Cooling Fan E-81 (1B-NNS)	20 A Breaker	20 A Breaker
226	AH-37 (1B-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
227	AH-38 (1B-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
228	AH-39 (1B-NNS) Motor Space Heater	15 A Breaker	15 A Breaker
229	CFC-AH-37 (1B-NNS)	1600 A Breaker with 200 A sensor	400 A Fuse
230	CFC-AH-38 (1B-NNS)	1600 A Breaker with 200 A sensor	400 A Fuse
231	CFC-AH-39 (1B-NNS)	1600 A Breaker with 200 A sensor	400 A Fuse
232	CRDM Fan E-80 (1B-NSS)	100 A Breaker	100 A Breaker
233	CRDM Fan E-81 (1B-NSS)	100 A Breaker	100 A Breaker
234	Reactor Coolant Drain Tank Pump 1B	50 A Breaker	50 A Breaker
235	Containment Building Sump Pump 1B-NNS	50 A Breaker	50 A Breaker

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
236.	Incore Instrument Drive Assemblies	15 A Breaker	15 A Breaker
237	MOV-1RC-V528SN-1 (8000C)	15 A Breaker	15 A Breaker
238	Lighting Panel LP-104	150 A Breaker at 120 v. A.C.	70 A Breaker at 480 v. A.C.
239	Lighting Panel LP-107 (N/E)	100 A Breaker at 120 v. A.C.	50 A Breaker at 480 v. A.C.
240	Lighting Panel LP-103	110 A Breaker at 120 v. A.C.	50 A Breaker at 480 v. A.C.
241	Lighting Panel LP-123	125 A Breaker at 120 v. A.C.	60 A Breaker at 480 v. A.C.
242	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
243	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
244	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
245	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
246	Power Receptacles #1-12, 1-16	60 A Breaker	60 A Breaker
247	Power Receptacles #1-3, 1-7	60 A Breaker	60 A Breaker
248	Power Receptacles #1-4, 1-8	60 A Breaker	60 A Breaker
249	RCP-1B-SN Oil Bearing Lift Pump	30 A Breaker	30 A Breaker
250	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
251	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
252	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
253	Pressurizer Heater Back-up Group "B"	90 A Breaker	100 A Fuse
254	Digital Rod Position Indication Cab "B" 120 V AC Supply	50 A Breaker	100 A Breaker
255	Power Receptacles #1-11, 1-15	60 A Breaker	60 A Breaker
256	Power Receptacles #1-77, 1-78	60 A Breaker	60 A Breaker
257	Stud Tensioner Hoist Motor (CRDM Terminal Box B1263)	15 A Breaker	15 A Breaker
258	RCP-1C-SN	Relay Trips Feeder Bkr.	Relay Trips Upstream Brk.
259	RCP-1C-SN	Relay Trips Feeder Bkr.	Relay Trips Upstream Brk.
260	Containment Fan Cooler AH-2 (1A-SA)	225 A Breaker	1600 A Breaker
261	Containment Fan Cooler AH-2 (1A-SA)	225 A Breaker	1600 A Breaker
262	Containment Fan Cooler AH-2 (1B-SA)	225 A Breaker	1600 A Breaker
263	Containment Fan Cooler AH-2 (1B-SA)	225 A Breaker	1600 A Breaker
264	Fan S-1 (1A-NNS) Filtration Unit MIS-1AR-7644	15 A Breaker	15 A Breaker
265	RCP-1C-SN Space Heater	15 A Breaker	20 A Breaker
266	Damper AR-D3-1 (Sol. Valve FSE-AR-D3-1)	6 A Fuse	15 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
267	Charcoal Temp. Detection Fan S-1 (1A-NNS)	6 A Fuse	15 A Breaker
268	Airborne Radioactivity Removal Unit S-1 (1A-NNS)	90 A Breaker	90 A Breaker
269	RCP-1C-SN Oil Bearing Lift Pump	30 A Breaker	30 A Breaker
270	Containment Building Sump Pump 1A-NNS	50 A Breaker	50 A Breaker
271	Airborne Radioactivity Removal Unit S-1 (1B-NNS)	90 A Breaker	90 A Breaker
272	Fuel Transfer Cont. Cab (Pump Motor)	15 A Breaker	15 A Breaker
273	RCC Change Fixt (Gripper Hoist Ratio Motor)	15 A Breaker	15 A Breaker
274	Fuel Transfer Manipulator Crane	30 A Breaker	30 A Breaker
275	RA-1CR-3584	0.6 A Fuse	20 A Breaker
276	RA-1CR-3585	0.6 A Fuse	15 A Breaker
277	RA-1CR-3586	0.6 A Fuse	15 A Breaker
278	RA-1CR-3587	0.6 A Fuse	15 A Breaker
279	2SI-V537 SA-1 Limit Switch	8 A Fuse	15 A Breaker
280	2MD-V36 SA-1 Limit Switch	8 A Fuse	15 A Breaker
281	Damper CV-D3 SA-1	6 A Fuse	20 A Breaker
282	AOV-2CP-B1 SA-1	6 A Fuse	20 A Breaker
283	Damper CV-D5 SA-1	6 A Fuse	20 A Breaker

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
284	RA-1CR-3575	0.6 A Fuse	15 A Breaker
285	RA-1CR-3576	0.6 A Fuse	15 A Breaker
286	RA-1CR-3577	0.6 A Fuse	15 A Breaker
287	RA-1CR-3582	0.6 A Fuse	20 A Breaker
288	RA-1CR-3583	0.6 A Fuse	20 A Breaker
289	2BD-F4SN-1 (PCV-8400B) Position Switches	6 A Fuse	15 A Breaker
290	2BD-F5SN-1 (PCV-8400C) Position Switches	6 A Fuse	15 A Breaker
291	MOV-2SI-535 SA-1 (8808C) ANN & Position Switches	3 A Fuse	15 A Breaker
292	LOCAL FIS-1AR-7647 A FAN S1 (1A-NNS) Flow Switch	6 A Fuse	15 A Breaker
293	LOCAL FS-7647B FAN S1 (1A-NNS) Flow Switch	6 A Fuse	15 A Breaker
294	Damper AR-D4-1 Limit Switch	6 A Fuse	15 A Breaker
295	Fire Detection Control Panel FAN S1 (1B-NNS)	6 A Fuse	15 A Breaker
296	Fuel Transfer Console Reactor Side 120/208 V Supply	20 A Breaker	20 A Breaker
297	Fire Detection Control Panel FAN S1 (1B-NSS)	20 A Breaker	20 A Breaker
298	Containment Fan Cooler AH-1 (1A-SB)	225 A Breaker	1600 A Breaker
299	Containment Fan Cooler AH-1 (1A-SB)	225 A Breaker	1600 A Breaker

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Item No.	Equipment Description	Primary Protection	Secondary Protection
300	Containment Fan Cooler AH-1 (1B-SB)	225 A Breaker	1600 A Breaker
301	Containment Fan Cooler AH-1 (1B-SB)	225 A Breaker	1600 A Breaker

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve given in Table 3.8-2 shall be bypassed only under accident conditions by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not capable of being bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The thermal overload protection for the above required valves shall be verified to be bypassed only under accident conditions by an OPERABLE integral bypass device by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry:

a. At least once per 18 months for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions; and

b. Following maintenance on the motor starter.

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TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION
1CS-341 (2CS-V522)	RCP A SEAL ISOL
1CS-382 (2CS-V523)	RCP B SEAL ISOL
1CS-423 (2CS-V524)	RCP C SEAL ISOL
1CS-182 (2CS-V600)	CSIP A MINIFLOW ISOLATION
1CS-210 (2CS-V601)	CSIP B MINIFLOW ISOLATION
1CS-196 (2CS-V602)	CSIP C MINIFLOW ISOLATION
1CS-235 (2CS-V609)	CSIP to RCS ISOLATION
1CS-166 (2CS-L521)	VCT ISOLATION
1CS-292 (2CS-L522)	RWST ISOLATION
1CS-214 (2CS-V585)	CSIPS MINIFLOW ISOLATION
1CS-165 (2CS-L520)	VCT ISOLATION
1CS-291 (2CS-L523)	RWST ISOLATION
1CS-238 (2CS-V610)	CSIP TO RCS ISOLATION
1CS-170 (2CS-V587)	CSIP SUCTION ISOLATION
1CS-169 (2CS-V589)	CSIP SUCTION ISOLATION
1CS-171 (2CS-V590)	CSIP SUCTION ISOLATION
1CS-168 (2CS-V588)	CSIP SUCTION ISOLATION
1CS-219 (2CS-V603)	CSIP DISCHARGE ISOL
1CS-217 (2CS-V604)	CSIP DISCHARGE ISOL
1CS-218 (2CS-V605)	CSIP DISCHARGE ISOL
1CS-220 (2CS-V606)	CSIP DISCHARGE ISOL
1CS-240 (2CS-V611)	SEAL WATER INJECTION
1CS-278 (2CS-V586)	BORIC ACID TANK TO CSIP
1CS-746 (2CS-V757)	CSIP MINIFLOW
1CS-752 (2CS-V759)	CSIP MINIFLOW
1CS-753 (2CS-V760)	CSIP MINIFLOW
1CS-745 (2CS-V758)	CSIP MINIFLOW
1CS-472 (2CS-V517)	RCP SEAL WATER RETURN ISOL
1CS-470 (2CS-V516)	RCP SEAL WATER ISOLATION
1RH-25 (2RH-V507)	RHR TO CSIP SUCTION
1RH-63 (2RH-V506)	RHR TO CSIP SUCTION
1RH-31 (2RH-F513)	RHR A MINI FLOW
1RH-69 (2RH-F512)	RHR B MINI FLOW
1RH-2 (1RH-V503)	RHRS INLET ISOLATION
1RH-40 (1RH-V501)	RHRS INLET ISOLATION
1RH-1 (1RH-V502)	RHRS INLET ISOLATION
1RH-39 (1RH-V500)	RHRS INLET ISOLATION
1SI-1 (2SI-V503)	BORON INJECTION TANK INLET ISOL
1SI-4 (2SI-V506)	BORON INJECTION TANK OUTLET ISOL
1SI-2 (2SI-V504)	BORON INJECTION TANK INLET ISOL
1SI-3 (2SI-V505)	BORON INJECTION TANK OUTLET ISOL
1SI-246 (2SI-V537)	ACCUMULATOR A DISCHARGE ISOLATION
1SI-248 (2SI-V535)	ACCUMULATOR C DISCHARGE ISOLATION
1SI-300 (2SI-V571)	CNMT SUMP TO RHR PUMP A ISOL
1SI-310 (2SI-V573)	CNMT SUMP TO RHR PUMP A ISOL
1SI-247 (2SI-V536)	ACCUM B DISCHARGE ISOLATION

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION
1SI-301 (2SI-V570)	CNMT SUMP TO RHR PUMP B ISOL
1SI-311 (2SI-V572)	CNMT SUMP TO RHR PUMP B ISOL
1SI-107 (2SI-V500)	HH SI TO RCS HL
1SI-52 (2SI-V502)	HH SI TO RCS CL
1SI-86 (2SI-V501)	HH SI TO RCS HL
1SI-326 (2SI-V577)	LH SI TO RCS HL
1SI-327 (2SI-V576)	LH SI TO RCS HL
1SI-340 (2SI-V579)	LH SI TO RCS CL
1SI-341 (2SI-V578)	LH SI TO RCS CL
1SI-359 (2SI-V587)	LH SI TO RCS HL
1SI-322 (2SI-V575)	RWST TO RHR A ISOL
1SI-323 (2SI-V574)	RWST TO RHR B ISOL
1CC-128 (3CC-B5)	CCS NONESSENTIAL RETURN ISOL
1CC-127 (3CC-B6)	CCS NONESSENTIAL RETURN ISOL
1CC-99 (3CC-B19)	CCS NONESSENTIAL RETURN ISOL
1CC-113 (3CC-B20)	CCS NONESSENTIAL RETURN ISOL
1CC-147 (3CC-V165)	RHR COOLING ISOL
1CC-167 (3CC-V167)	RHR COOLING ISOL
1CC-176 (2CC-V172)	CVCS HX CNMT ISOLATION
1CC-202 (2CC-V182)	CVCS HX CNMT ISOLATION
1CC-208 (2CC-V170)	CCW-RCPS ISOLATION
1CC-299 (2CC-V183)	RCPS BEARING HX ISOLATION
1CC-251 (2CC-V190)	RCPS THER BARRIER ISOLATION
1CC-207 (2CC-V169)	CCW-RCPS ISOLATION
1CC-297 (2CC-V184)	RCPS BEARING HX ISOLATION
1CC-249 (2CC-V191)	RCPS THER BARRIER ISOLATION
1CT-105 (2CT-V6)	CNMT SPRAY SUMP A RECIRC ISOL
1CT-102 (2CT-V7)	CNMT SPRAY SUMP B RECIRC ISOL
1CT-26 (2CT-V2)	CNMT SPRAY PUMP A INJECT. SUPPLY
1CT-71 (2CT-V3)	CNMT SPRAY PUMP B INJECT. SUPPLY
1CT-50 (2CT-V21)	SPRAY HDR A ISOLATION
1CT-12 (3CT-V85)	NAOH ADDITIVE ISOLATION
ICT-88 (2CT-V43)	SPRAY HDR B ISOLATION
ICT-11 (3CT-V88)	NAOH ADDITIVE ISOLATION
1CT-47 (2CT-V25)	CNMT SPRAY HDR A RECIRC
1CT-24 (2CT-V8)	CNMT SPRAY PUMP A EDUCTOR TEST
1CT-95 (2CT-V49)	CNMT SPRAY HDR B RECIRC
1CT-25 (2CT-V145)	CNMT SPRAY PUMP B EDUCTOR TEST
1AF-5 (3AF-V187)	AFWP A RECIRC
1AF-24 (3AF-V188)	AFWP B RECIRC
1AF-55 (2AF-V10)	AFW TO SG A ISOL*
1AF-93 (2AF-V19)	AFW TO SG B ISOL*
1AF-74 (2AF-V23)	AFW TO SG C ISOL*
1AF-137 (2AF-V116)	AFWTD TO SG A ISOL*
1AF-143 (2AF-V117)	AFWTD TO SG B ISOL*
1AF-149 (2AF-V118)	AFWTD TO SG C ISOL*
1MS-70 (2MS-V8)	AFWTD STEAM B ISOLATION*

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION
1MS-72 (2MS-V9)	AFWTD STEAM C ISOLATION
1SW-39 (3SW-B5)	NORMAL SW HDR A ISOLATION
1SW-276 (3SW-B8)	NORMAL SW HDR A RETURN ISOL
1SW-270 (3SW-B15)	SW HDR A TO AUX RSVR ISOL
1SW-40 (3SW-B6)	NORMAL SW HDR B ISOL
1SW-275 (3SW-B13)	SW HDR A RETURN ISOL
1SW-274 (3SW-B14)	SW HDR B RETURN ISOL
1SW-271 (3SW-B16)	SW HDR B TO AUX RSVR ISOL
1SW-92 (2SW-B46)	SW TO FAN CLR AH3 INLET
1SW-97 (2SW-B47)	SW FROM FAN CLR AH3 OUTLET
1SW-91 (2SW-B45)	SW TO FAN CLR AH2 INLET
1SW-109 (2SW-B49)	SW FROM FAN CLR AH2 OUTLET
1SW-225 (2SW-B52)	SW TO FAN CLR AH1 INLET
1SW-98 (2SW-B48)	SW FROM FAN CLR AH1 OUTLET
1SW-227 (2SW-B51)	SW TO FAN CLR AH4 INLET
1SW-110 (2SW-B50)	SW FROM FAN CLR AH4 OUTLET
1SW-124 (3SW-B70)	SW TO AFWTD PUMP
1SW-126 (3SW-B71)	SW TO AFWTD PUMP
1SW-129 (3SW-B73)	SW TO AFWTD PUMP
1SW-127 (3SW-B72)	SW TO AFWTD PUMP
1SW-123 (3SW-B75)	SW TO AFW PUMP A SUPPLY
1SW-121 (3SW-B74)	SW TO AFW PUMP A SUPPLY
1SW-132 (3SW-B77)	SW TO AFW PUMP B SUPPLY
1SW-130 (3SW-B76)	SW TO AFW PUMP B SUPPLY
1ED-94 (2MD-V36)	CNMT SUMP ISOLATION
1ED-95 (2MD-V77)	CNMT SUMP ISOLATION
3CZ-B5	RAB ELEC PROT INLET
3CZ-B6	RAB ELEC PROT INLET
3CZ-B7	RAB ELEC PROT EXHAUST
3CZ-B8	RAB ELEC PROT EXHAUST
3CZ-B32	RAB ELEC PROT PURGE MAKE-UP
3CZ-B33	RAB ELEC PROT PURGE MAKE-UP
3CZ-B34	RAB ELEC PROT PURGE INLET
3CZ-B35	RAB ELEC PROT PURGE INLET
3FV-B2	FUEL HANDLING EXHAUST INLET*
3FV-B4	FUEL HANDLING EXHAUST INLET*
3CZ-B1	CONTROL ROOM NORMAL SUPPLY ISOL*
3CZ-B3	CONTROL ROOM NORMAL EXHAUST ISOL*
3CZ-B17	CONTROL ROOM PURGE MAKE UP*
3CZ-B2	CONTROL ROOM NORMAL SUPPLY ISOL*
3CZ-B4	CONTROL ROOM EXHAUST ISOLATION*
3CZ-B18	CONTROL ROOM PURGE MAKE UP*
3CZ-B14	CONTROL ROOM PURGE EXHAUST*

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TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	FUNCTION
3CZ-B26	CONTROL ROOM NORMAL SUPPLY DISCH*
3CZ-B25	CONTROL ROOM SUPPLY DISCHARGE*
3CZ-B13	CONTROL ROOM PURGE EXHAUST*
3CZ-B12	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
3CZ-B10	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
3CZ-B9	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
3CZ-B11	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
3CZ-B23	CONTROL ROOM EMER FLTR INLET*
3CZ-B21	CONTROL ROOM FLTR DISCHARGE*
3CZ-B22	CONTROL ROOM EMER FLTR DISCHARGE*
3CZ-B24	CONTROL ROOM EMER FLTR INLET*
3CZ-B19	CONTROL ROOM EMER FLTR DISCHARGE*
3CZ-B20	CONTROL ROOM EMER FLTR DISCHARGE*
3AV-B1	RAB EMER EXHAUST INLET
3AV-B2	RAB EMER EXHAUST OUTLET
3AV-B4	RAB EMER EXHAUST INLET
3AV-B5	RAB EMER EXHAUST OUTLET
3AV-B3	RAB EMER EXHAUST BLEED
3AV-B6	RAB EMER EXHAUST BLEED
3AC-B2	RAB SWGR B EXHAUST
3AC-B3	RAB SWGR B EXHAUST
3AC-B1	RAB SWGR A EXHAUST

*Overload bypass for these valves is accomplished by the activation slave relays in the circuit. These activation slave relays are tested as a part of the Engineered Safety Features Actuation System instrumentation in accordance with the requirements of Table 4.3-2.

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TABLE 4.9-1

ADMINISTRATIVE CONTROLS TO PREVENT DILUTION DURING REFUELING

VALVE/ID	VALVE POSITION DURING REFUELING	LOCK	DESCRIPTION
1CS-149 (CS-D121SN)	Closed	Yes	RMW to the CVCS makeup control system
1CS-510 (CS-D631SN)	Closed	Yes	Boric Acid Batch Tank Outlet valve. May be opened if the batching tank concentration is $\geq$ 2000 ppm boron, and valve 1CS-503 (makeup water supply to batch tank) is closed.
1CS-503 (CS-D251)	Closed	Yes	RMW to Batching Tank. Do not open unless outlet valve 1CS-510 is closed.
1CS-570 (CS-D575SN)	Closed	No	CVCS letdown to BTRS. Place valve in "shut" at valve control switch and place BTRS function selector switch in "off." No lock required.
1CS-670 (CS-D599SN)	Closed	Yes	RMW to BTRS loop.
1CS-649 (CS-D198SN)	Closed	Yes	Resin sluice to BTRS demineralizers.
1CS-93 (CS-D51SN)	Closed	Yes	Resin sluice to CVCS demineralizers
1CS-320 (CS-D641SN)	Closed	Yes	Recycle Evaporation Feed Pump to charging/safety injection pump suction.
1CS-98 (CS-D740SN)	Open	No	BTRS bypass valve. Place valve control switch in "open" position.

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3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine, used for movement of fuel assemblies, having:

   1. A minimum capacity of 4000 pounds, and

   2. An automatic overload cutoff limit less than or equal to 2700 pounds.

b. The auxiliary hoist, used for latching and unlatching drive rods, having:

   1. A minimum capacity of 3000 pounds, and

   2. A 1000-pound load indicator that shall be used to monitor loads to prevent lifting more than 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for the refueling machine and/or auxiliary hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE, within 100 hours prior to the start of such operations, by performing a load test of at least 4000 pounds and demonstrating an automatic load cutoff at less than or equal to 2700 pounds.

4.9.6.2 The auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 900 pounds.

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3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Emergency Exhaust System Trains shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in a storage pool.

ACTION:

a. With one Fuel Handling Building Emergency Exhaust System Train inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Handling Building Emergency Exhaust System Train is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorber.

b. With no Fuel Handling Building Emergency Exhaust System Trains OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Handling Building Emergency Exhaust System Train is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Emergency Exhaust System trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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SURVEILLANCE REQUIREMENTS (Continued)

4.9.12 (Continued)

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is not greater than 4.1 inches water gauge while operating the unit at a flow rate of 6600 cfm ± 10%,

2. Verifying that, on a High Radiation test signal, the system automatically starts and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch water gauge, relative to the outside atmosphere, during system operation at a flow rate of 6600 cfm ± 10%,

4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

5. Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the unit at a flow rate of 6600 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.12 (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6600 cfm ± 10%.

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases:

a. Outside temporary tank, excluding demineralizer vessels and liners used to solidify or to dewater radioactive wastes.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents within 7 days following any addition of radioactive material to the tank.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge or Vent	P Each PURGE(3) Grab Sample	P Each PURGE(3)	Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. a. Plant Vent Stack	M(3),(4),(5) Grab Sample		Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
b. Turbine Bldg Vent Stack, Waste Processing Bldg Vent Stacks 5&5A	M Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
			H-3 (oxide) (Turbine Bldg. Vent Stack)	1×10^{-6}
4. All Release Types as listed in 1., 2., and 3. above	Continuous(6)	W(7) Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous(6)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(6)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(6)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose, from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit the the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

FINAL DRAFT

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1770 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal, but a 2000 pcm SHUTDOWN MARGIN is required to provide adequate protection for postulated inadvertent dilution events. The unit "pcm" is used throughout these specifications to conform with the reactivity information provided by the NSSS supplier; 1000 pcm is equal to 1% $\Delta k/k$.

Analysis of inadvertent boron dilution at cold shutdown is based on:

1. all RCCA's in the core while the RCS, except the reactor vessel, is drained (i.e., not filled), and
2. all RCCA's, except shutdown banks C and D, are fully inserted in the core while the RCS is filled.

In addition, by assuming the most reactive control rod is stuck out of the core, its worth is effectively added to the 2000 pcm shutdown margin in calculating the necessary soluble boron concentration.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; i.e., the positive limit is based on core conditions for all rods withdrawn, BOL, hot zero THERMAL POWER, and the negative limit is based on core conditions for all rods withdrawn, EOL, RATED THERMAL POWER. Accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -42 pcm/°F. The MTC value of -33 pcm/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -42 pcm/°F.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging/safety injection pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1770 pcm after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16800 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 436,000 gallons of 2000-2200 ppm borated water be maintained in the refueling water storage tank (RWST).

With the RCS temperature below 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 335°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1000 pcm after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4900 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 82,000 gallons of 2000-2200 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified value, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values. The specified percent level and gallons differ by less than 0.1%.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 AND 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOW RATE, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

BASES

3/4.3.1 AND 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. For example, if a bistable has a trip setpoint of $\leq$100%, a span of 125%, and a calibration accuracy of ±0.50%, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is $\leq$100.62%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties,in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured"

REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor draft factor, an increased rack drift factor, and provides a threshold value for determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) charging/safety injection pumps start and automatic valves position, (2) reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment fan coolers start and automatic valves position, (11) emergency service water pumps start and automatic valves position, and (12) control room isolation and emergency filtration start.

BASES

REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and low steam-line pressure, sends an open signal to the accumulator discharge valves and automatically blocks steam-line isolation on a high rate of decrease in steam-line pressure. On decreasing pressurizer pressure, P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steam-line pressure and allows steam-line isolation, on a high rate of decrease in steam-line pressure, to become active upon manual block of Safety Injection from low steam-line pressure.

P-12 P-12 has no ESF or reactor trip functions. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of emergency systems.

INSTRUMENTATION

BASES

REMOTE SHUTDOWN SYSTEM (Continued)

This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 DELETED

3/4.3.3.9 METAL IMPACT MONITORING SYSTEM

The OPERABILITY of the Metal Impact Monitoring System ensures that sufficient capability is available to detect loose metallic parts in the Reactor System and avoid or mitigate damage to Reactor System components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE TREATMENT SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} μCi/ml are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and 10 CFR 50 Appendix G. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1). The Shearon Harris Unit 1 cooldown and heatup limitations shown in Figures 3.4-2 and 3.4-3 and Table 4.4-6 are not impacted by the 120°F limit.

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

PRESSURE/TEMPERATURE LIMITS (Continued)

Copper Trend Curves shown in Figure B 3/4.4-2. The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 and Table 4.4-6 include predicted adjustments for this shift in RT_{NDT} at the end of 4 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

assumed cannot occur, Technical Specifications require lockout of all but one charging/safety injection pump while in MODES 4, 5, and 6 (below 335°F) with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition and Addenda through Summer 1978.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a, during performance of the periodic test, to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of -2 psig, and (2) the containment peak pressure does not exceed the design pressure of 45 psig.

The maximum peak pressure expected to be obtained from a postulated main steam line break event is 40.9 psig using a value of 1.9 psig for initial positive containment pressure. However, since the instrument tolerance for containment pressure is 1.32 psig and the high-one setpoint is 3.0 psig, the pressure limit was reduced from the high-one setpoint by slightly more than the tolerance and was set at 1.6 psig. This value will prevent spurious safety injection signals caused by instrument drift during normal operation. The -1" wg was chosen to be consistent with the initial assumptions of the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 41 psig in the event of a postulated main steam line break accident. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment preentry purge makeup and exhaust isolation valves are required to be sealed closed during plant operations in MODES 1, 2, 3 and 4 since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during these MODES ensures that excessive quantities of radioactive materials will not be released via the Pre-entry Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Normal Containment Purge System is restricted to the 8-inch purge makeup and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during normal containment PURGING operation. The total time the Normal Containment Purge System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4.

Leakage integrity tests with a maximum allowable leakage rate for containment purge makeup and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before

CONTAINMENT VENTILATION SYSTEM (Continued)

gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Fan Coolers are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses. The RWST level of 436,000 gallons provides adequate test conditions to demonstrate that the flow rate is within the maximum and minimum assumptions of the analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Fan Coolers ensures that adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Fan Coolers and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere.

As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Fan Coolers have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. This hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," Rev. 2, November 1978.

3/4.6.5 VACUUM RELIEF SYSTEM

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than -1.93 psig. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of -2 psig.

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

The OPERABILITY of the Emergency Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," Rev. 2, January 1976.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the Control Room Emergency Filtration System ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon a removal efficiency of 99% for elemental, particulate and organic forms of radioiodine. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

3/4.7.7 REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEM

The OPERABILITY of the Reactor Auxiliary Building Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the

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REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEM (Continued)

pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Manager-Technical Support. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. In order to establish the inspection frequency for each type of snubber, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next

SNUBBERS (Continued)

inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.9 SEALED SOURCE CONTAMINATION

The sources requiring leak tests are specified in 10 CFR 31.5(c)(2)(ii). The limitation on removable contamination is required by 10 CFR 31.5(c)5. This

SEALED SOURCE CONTAMINATION (Continued)

limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources that are frequently handled are required to be tested more often than those that are not. Sealed sources that are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 DELETED

3/4.7.11 DELETED

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits do not include an allowance for instrument errors.

3.4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

The OPERABILITY of the Emergency Service Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.8.1, 3/4.8.2, AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The switchyard is designed using a breaker-and-a-half scheme. The switchyard currently has six connections with the CP&L transmission network; each of these transmission lines is physically independent. The switchyard has one connection with each of the two Startup Auxiliary Transformers and each SAT can be fed directly from an associated offsite transmission line. The Startup Auxiliary Transformers are the preferred power source for the Class 1E ESF buses. The minimum alignment of offsite power sources will be maintained such that at least two physically independent offsite circuits are available. The two physically independent circuits may consist of any two of the incoming transmission lines to the SATs (either through the switchyard or directly) and into the Class 1E system. As long as there are at least two transmission lines in service and two circuits through the SATs to the Class 1E buses, the LCO is met.

During MODES 5 and 6, the Class 1E buses can be energized from the offsite transmission net work via a combination of the main transformers, and unit auxiliary transformers. This arrangement may be used to satisfy the requirement of one physically independent circuit.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

BASES

A.C. SOURCES, D.C. SOURCES, AMD ONSITE POWER DISTRIBUTION (Continued)

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based upon the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 as modified in accordance with the guidance of IE Notice 85-32, April 22, 1985; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979. The inclusion of the loss of generator potential transformer circuit lockout trip is a design feature based upon coincident logic and is an anticipatory trip prior to diesel generator overspeed.

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not

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A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The bypassing of the motor-operated valves thermal overload protection during accident conditions by integral bypass devices ensures that safety-related valves will not be prevented from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection during accident conditions are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND NEW AND SPENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

The limitations on the Fuel Handling Building Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Criteria for laboratory testing of charcoal and for in-place testing of HEPA filters and charcoal adsorbers is based upon removal efficiencies of 95% for organic and elemental forms of radioiodine and 99% for particulate forms. The filter pressure drop was chosen to be half-way between the estimated clean and dirty pressure drops for these components. This assures the full functionality of the filters for a prolonged period, even at the Technical Specification limit.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	200 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential greater than 320°F but less than 625°F.
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to ≥ 1481 psig.

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6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant General Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Foreman (or, during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President-Harris Nuclear Project shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. An individual qualified as a Radiation Control Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. (Deleted).

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*The Radiation Control Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

CORPORATE ORGANIZATION

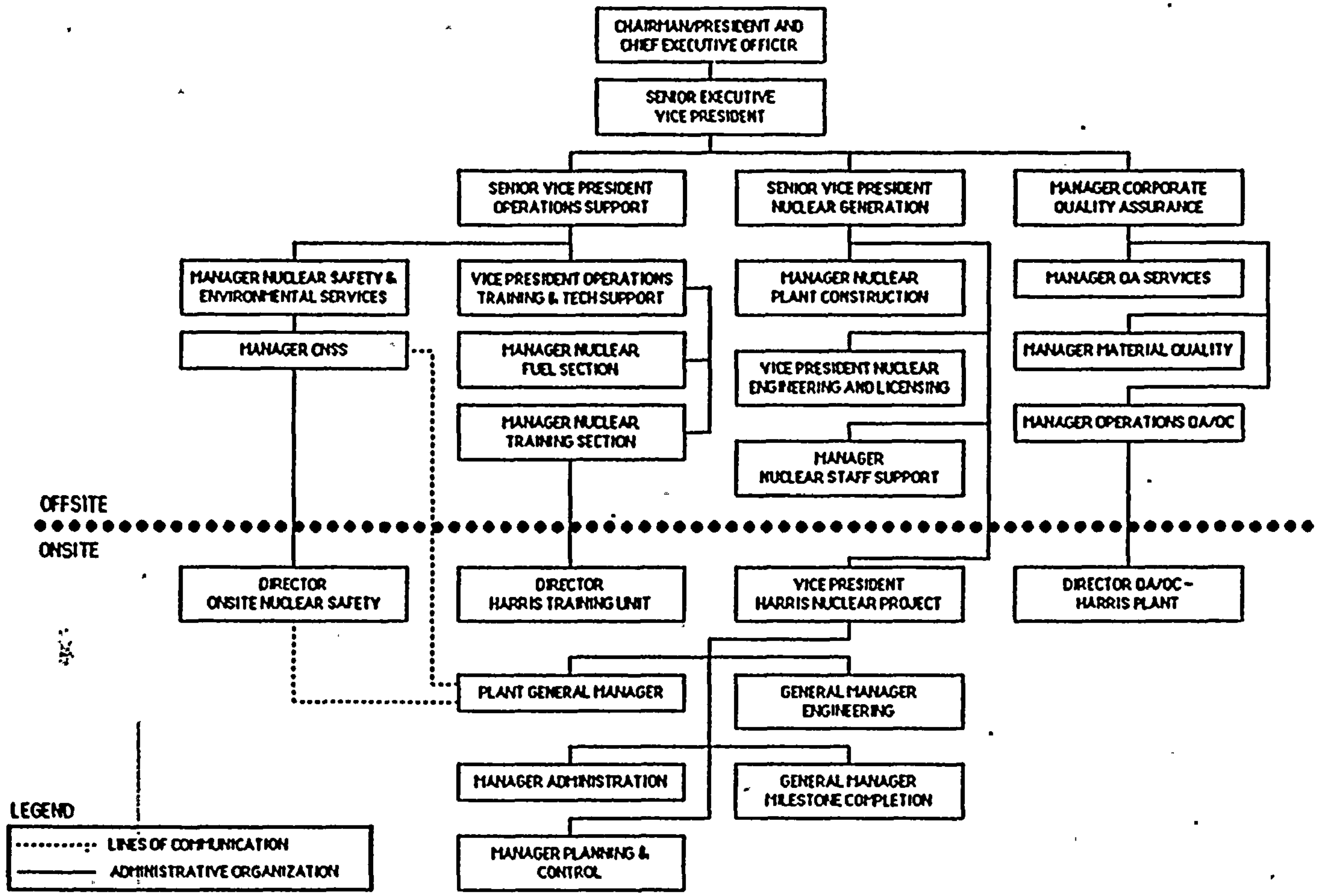

FIGURE 6.2-1

OFFSITE ORGANIZATION

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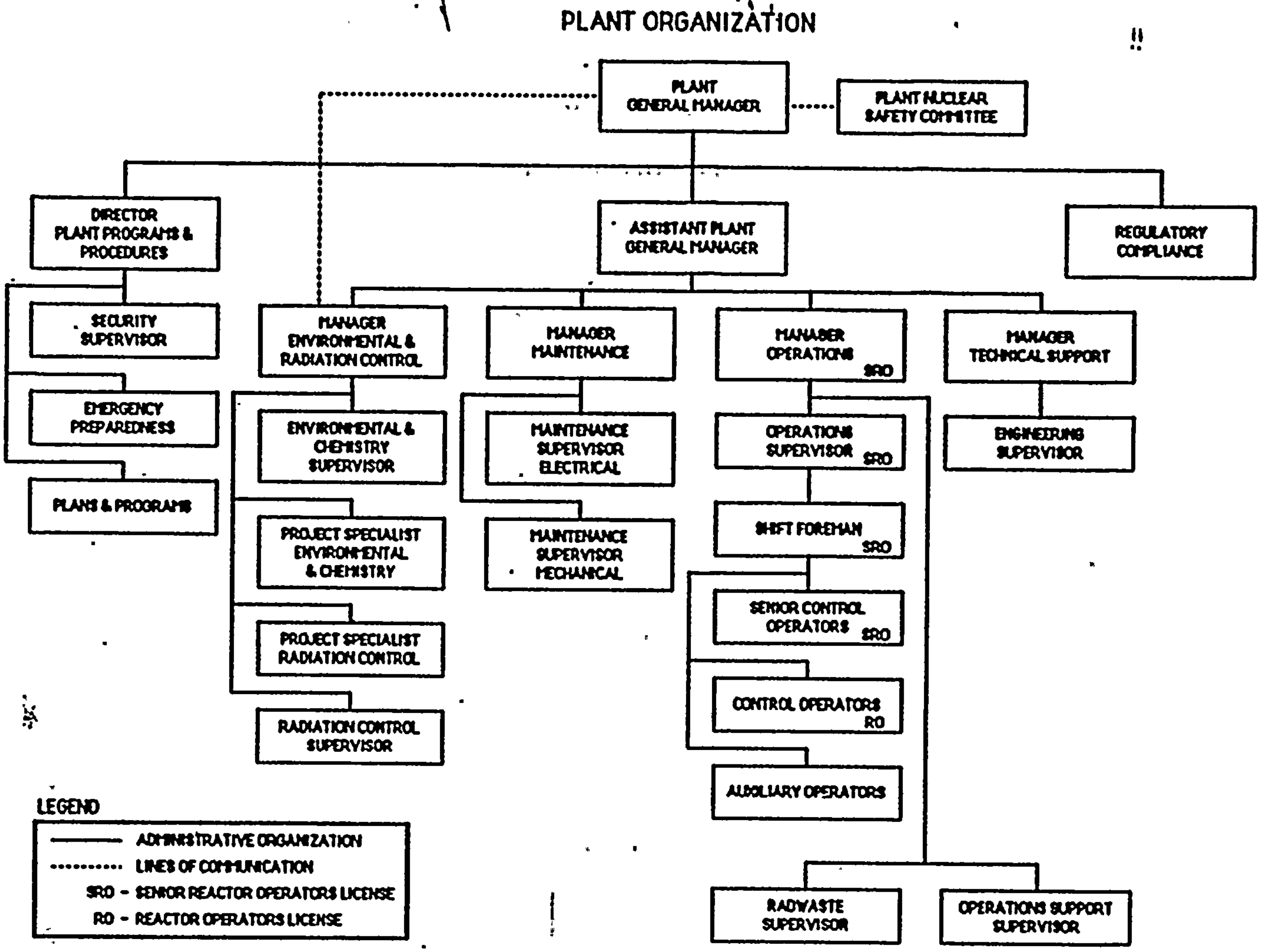

FIGURE 6.2-2

UNIT ORGANIZATION

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6.2.3 ONSITE NUCLEAR SAFETY (ONS) UNIT

FUNCTION

6.2.3.1 The ONS Unit shall function to examine unit operating characteristics, NRC issuances, industry advisories (including information forwarded by INPO from their evaluation of all industry LERs), and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety, to appropriate levels of management, up to and including the Senior Vice President-Operations Support, if necessary.

COMPOSITION

6.2.3.2 The ONS Unit shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a baccalaureate degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ONS Unit shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ONS Unit shall be prepared, maintained, and forwarded each calendar month to the Manager-Nuclear Safety and Environmental Services.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Foreman in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a baccalaureate degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of the September 1979 draft of ANS 3.1, with the exceptions and alternatives noted on FSAR pages 1.8-8 (Am.20), 1.8-9 (Am.17), 1.8-10 (Am.22),

*Not responsible for sign-off function.

UNIT STAFF QUALIFICATIONS (Continued

1.8-11 (Am.20), 1.8-12 (Am.17), and 1.8-13 (Am.17), for comparable positions, except for the Manager-Environmental and Radiation Control who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclsure 1 of the March 28, 1980, NRC letter to all licensees.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director-Harris Training Unit and shall meet or exceed the requirements and recommendations of the September 1979 draft of ANS 3.1, with the exceptions and alternatives noted on FSAR pages 1.8-8 (Am.20), 1.8-9 (Am.17), 1.8-10 (Am.22), 1.8-11 (Am.20), 1.8-12 (Am.17), and 1.8-13 (Am.17), and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 SAFETY AND TECHNICAL REVIEWS

6.5.1.1 General Program Control

6.5.1.1.1 A safety and a technical evaluation shall be prepared for each of the following:

a. All procedures and programs required by Specification 6.8, other procedures that affect nuclear safety, and changes thereto;

b. All proposed tests and experiments that are not described in the Final Safety Analysis Report; and

c. All proposed changes or modifications to plant systems or equipment that affect nuclear safety.

6.5.1.2 Technical Evaluations

6.5.1.2.1 Technical evaluations will be performed by personnel qualified in the subject matter and will determine the technical adequacy and accuracy of the proposed activity. If interdisciplinary evaluations are required to cover the technical scope of an activity, they will be performed.

6.5.1.2.2 Technical review personnel will be identified by the responsible Manager or his designee for a specific activity when the review process begins.

6.5.1.3 Qualified Safety Reviewers

6.5.1.3.1 The Plant General Manager shall designate those individuals who will be responsible for performing safety reviews described in Specification 6.5.1.4.

RESPONSIBILITIES (Continued)

b. Provide written notification within 24 hours to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services of disagreement between the PNSC and the Plant General Manager. However, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.2.8 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President-Harris Nuclear Project and the Manager-Nuclear Safety and Environmental Services.

6.5.3 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.3.1 The Corporate Nuclear Safety Section (CNSS) of the Nuclear Safety and Environmental Services Department shall function to provide independent review of plant changes, tests, and procedures; verify that REPORTABLE EVENTS are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer. They shall also evaluate all CP&L LERs for their potential applicability to other CP&L nuclear power plants.

ORGANIZATION

6.5.3.2 The individuals assigned responsibility for independent reviews shall be technically qualified in a specified technical discipline or disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Administrative controls,
i. Quality assurance practices,
j. Nondestructive testing, and
k. Other appropriate fields associated with the unique characteristics.

REVIEW (Continued)

e. Violations, which require written notification to the Commission, of applicable codes, regulations, orders, Technical Specifications, license requirements, internal procedures or instructions having nuclear safety significance, significant operating abnormalities or deviations from normal and expected performance of plant safety-related structures, systems, or components;

f. All REPORTABLE EVENTS;

g. All proposed modifications that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50 or involve a change to the Technical Specifications;

h. Any other matter involving safe operation of the nuclear power plant that the Manager-Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager-Corporate Nuclear Safety Section by the onsite operating organization or other functional organizational units within Carolina Power & Light Company;

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

j. Reports and minutes of the PNSC.

6.5.3.10 Review of items considered under Specification 6.5.3.9.e, h and j above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.3.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNSS files.

b. Recommendations and concerns shall be submitted to the Plant General Manager and Vice President-Harris Nuclear Project within 14 days of completion of the review. A report summarizing the reviews encompassed by Specification 6.5.3.9 shall be provided to the Plant General Manager and the Vice President-Harris Nuclear Project every other month.

c. A summation of recommendations and concerns of the Corporate Nuclear Safety Section shall be submitted to the Chairman/President and Chief Executive Officer and other appropriate senior management personnel at least every other month.

FINAL DRAFT

RECORDS

6.5.4.3 Records of audits shall be prepared and retained.

6.5.4.4 Audit reports encompassed by Specification 6.5.4.1 shall be prepared, approved by the Manager-Quality Assurance Services, and forwarded, within 30 days after completion of the audit, to the Senior Executive Vice President-Power Supply and Engineering and Construction, Senior Vice President-Nuclear Generation, Vice President-Harris Nuclear Project, Manager-Nuclear Safety and Environmental Services, Plant General Manager, and the management positions responsible for the areas audited.

AUTHORITY

6.5.4.5 The Manager-Quality Assurance Service Section, under the Manager-Corporate Quality Assurance Department, shall be responsible for the following:

a. Administering the Corporate Quality Assurance Audit Program.

b. Approval of the individuals selected to conduct quality assurance audits.

6.5.4.6 Audit personnel shall be independent of the area audited.

6.5.4.7 Selection of personnel for auditing assignments shall be based on experience or training that establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting audit personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

6.5.4.8 Qualified outside consultants, or other individuals independent from those personnel directly involved in plant operation, shall be used to augment the audit teams when necessary.

6.5.5 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.5.1 An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months using either qualified offsite licensee personnel or an outside fire protection firm.

6.5.5.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

6.5.5.3 Copies of the audit reports and responses to them shall be forwarded to the Vice President-Harris Nuclear Project and the Manager-Corporate Quality Assurance.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

PROCEDURES AND PROGRAMS (Continued)

g. Quality Assurance Program for effluent and environmental monitoring; and

h. Fire protection program implementation.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved in accordance with Specification 6.5.1 prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed in accordance with Specification 6.5.1, and approved within 14 days of implementation by the Plant General Manager or by the Manager of the functional area affected by the procedure.

6.8.4 The following programs shall be established, implemented, and maintained:

a. <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage, to as low as practical levels, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include:

1. Residual Heat Removal System and Containment Spray System, except spray additive subsystem and RWST,

2. Safety Injection System, except boron injection recirculation subsystem and accumulator,

3. Portions of the Chemical and Volume Control System:

   a. Letdown subsystem, including demineralizers,
   b. Boron re-cycle holdup tanks, and
   c. Charging/safety injection pumps,

4. Post-Accident Sample System,

PROCEDURES AND PROGRAMS (Continued)

a. Primary Coolant Sources Outside Containment (Continued)

5. Post-Accident Reactor Auxiliary Building Ventilation System,

6. Valve Leakoff Equipment Drain System,

7. Gaseous Waste Processing System, and

8. Seal Water Return System.

The program shall include (1) preventive maintenance and periodic visual inspection requirements and (2) integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program that will ensure the capability to determine accurately the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical variables and the control points for these variables,

2. Identification of the procedures used to measure the values of the critical variables,

3. Identification of process sampling points, which shall include monitoring for evidence of condenser in-leakage,

4. Procedures for the recording and management of data,

5. Procedures defining corrective actions for all off-control point chemistry conditions, and

ANNUAL REPORTS (Continued)

film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. The results of specific activity analyses in which the reactor coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) the time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

c. Documentation of all challenges to the pressurizer power-operated relief valves (PORVs) and safety valves.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.3 Routine Annual Radiological Environmental Operating Reports, covering the operation of the unit during the previous calendar year, shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls, as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.