December 6, 1984

Docket No. 50-458

DISTRIBUTION
Docket File
NRC PDR
TSRF File
LPDR
NRC Participants
EButcher
DHouston

LICENSEE: Gulf States Utilities Company

FACILITY: River Bend, Unit 1

SUBJECT: SUMMARY OF SITE VISIT ON NOVEMBER 1 AND 2, 1984 TO DISCUSS TECHNICAL SPECIFICATIONS

On November 1 and 2, 1984, the staff visited the River Bend site to tour the facility and to discuss the developement of technical specifications for the plant. The site tour was conducted by personnel from GSU. The technical specification discussion was attended by representatives of NRC, GSU and GSU consultants. The attendee lists for both days are enclosed (Enclosures 1 and 2). A copy of a GSU handout used for the technical specification meeting is also enclosed (Enclosure 3) plus replacement pages for their draft of River Bend Technical Specifications (Enclosure 4).

On the first day, the tour included the reactor building, auxiliary building, radwaste building, control building, diesel generator building, ultimate heat sink, turbine building, fire protection pump house, fuel building and the cooling tower area. During the lunch break, the group met with T. Plunkett (Plant Mgr.) to discuss the nature of the visit and the anticipated schedules for the plant completion and licensing actions. On the second day, the group briefly toured the training center which contains the EOF and plant simulator.

The process (at GSU) for the development of River Bend Technical Specifications was discussed which showed the interface between the utility and the A-E, NSSS and consultants and the procedures for development, review and resolution of concerns (Enclosure 3). The utility utilized Nuclear Energy Services, Inc. (NES) as a consultant to generate their first draft of technical specifications. This draft also had input from the A-E (Stone and Webster) and the NSSS vendor (General Electric). Within GSU, the development and review of technical specifications is the responsibility of a Reactor Engineer Supervisor who is also a licensed SRO on River Bend. At the conclusion of the meeting, GSU provided some marked up pages of River Bend Technical Specifications (Enclosure 4) which are replacement pages to be inserted in their submittal of July 17, 1984.

ORIGINAL SIGNED BY
M. Dean Houston, Project Manager
Technical Specifications Review Group
Division of Licensing

See next page

DL:TSRG	DL:TSRG
DHouston:jc	EButcher
12/5/84	12/ /84

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

December 6, 1984

Docket No. 50-458

LICENSEE: Gulf States Utilities Company

FACILITY: River Bend, Unit 1

SUBJECT: SUMMARY OF SITE VISIT ON NOVEMBER 1 AND 2, 1984 TO DISCUSS TECHNICAL SPECIFICATIONS

On November 1 and 2, 1984, the staff visited the River Bend site to tour the facility and to discuss the developement of technical specifications for the plant. The site tour was conducted by personnel from GSU. The technical specification discussion was attended by representatives of NRC, GSU and GSU consultants. The attendee lists for both days are enclosed (Enclosures 1 and 2). A copy of a GSU handout used for the technical specification meeting is also enclosed (Enclosure 3) plus replacement pages for their draft of River Bend Technical Specifications (Enclosure 4).

On the first day, the tour included the reactor building, auxiliary building, radwaste building, control building, diesel generator building, ultimate heat sink, turbine building, fire protection pump house, fuel building and the cooling tower area. During the lunch break, the group met with T. Plunkett (Plant Mgr.) to discuss the nature of the visit and the anticipated schedules for the plant completion and licensing actions. On the second day, the group briefly toured the training center which contains the EOF and plant simulator.

The process (at GSU) for the development of River Bend Technical Specifications was discussed which showed the interface between the utility and the A-E, NSSS and consultants and the procedures for development, review and resolution of concerns (Enclosure 3). The utility utilized Nuclear Energy Services, Inc. (NES) as a consultant to generate their first draft of technical specifications. This draft also had input from the A-E (Stone and Webster) and the NSSS vendor (General Electric). Within GSU, the development and review of technical specifications is the responsibility of a Reactor Engineer Supervisor who is also a licensed SRO on River Bend. At the conclusion of the meeting, GSU provided some marked up pages of River Bend Technical Specifications (Enclosure 4) which are replacement pages to be inserted in their submittal of July 17, 1984.

M. Dean Houston

M. Dean Houston, Project Manager
Technical Specifications Review Group
Division of Licensing

Enclosure:
As stated

CC: See next page

River Bend Station

Mr. William J. Cahill, Jr.
Senior Vice President
River Bend Nuclear Group
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704
ATTN: Mr. J. E. Booker

cc: Troy B. Conner, Jr., Esq.
Conner and Wetterhahn
1747 Pennsylvania Avenue, N. W.
Washington, D.C. 20006

Mr. William J. Reed, Jr.
Director - Nuclear Licensing
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704

H. Anne Plettinger
3456 Villa Rose Dr.
Baton Rouge, Louisiana 70806

Richard M. Troy, Jr., Esq.
Assistant Attorney General in Charge
State of Louisiana Department of Justice
234 Loyola Avenue
New Orleans, Louisiana, 70112

Dwight D. Chamberlain
Resident Inspector
Post Office Box 1051
St. Francisville, Louisiana 70775

Gretchen R. Rothschild
Louisianians for Safe Energy, Inc.
1659 Glenmore Avenue
Baton Rouge, Louisiana 70775

James W. Pierce, Jr., Esq.
P. O. Box 23571
Baton Rouge, Louisiana 70893

Ms. Linda B. Watkins/Mr. Steven Irving
Attorney at Law
355 Napoleon Street
Baton Rouge, Louisiana 70802

Mr. David Zaloudek
Nuclear Energy Division
Louisiana Department of
Environmental Quality
Post Office Box 14690
Baton Rouge, Louisiana 70898

Mr. J. David McNeill, III
William G. Davis, Esq.
Department of Justice
Attorney General's Office
7434 Perkins Road
Baton Rouge, Louisiana 70808

Enclosure 1

Site Tour Attendees

Main Plant Tour

November 1, 1984

NRC

S. Brown
J. Flores
D. Houston
C. Schulten
E. Weinkam

GSU

T. Overlied (Tour Guide)
D. Williamson
J. Price
B. Reed*
P. Graham*
R. Trowbridge*
T. Plunkett (Plant Mgr.)*

*-Partial Participation

Training Center Tour

November 2, 1984

NRC

S. Brown
D. Chamberlain (SRI)
J. Flores
D. Houston
E. Weinkam
C. Schulten

GSU

B. Reed
J. Price
D. Williamson
B. Odell

GSU Consultant

B. Price
C. Alm

Enclosure 2

Attendees

Technical Specification Meeting

November 2, 1984

NRC

S. Brown
J. Flores
D. Houston
C. Schulten
E. Weinkam

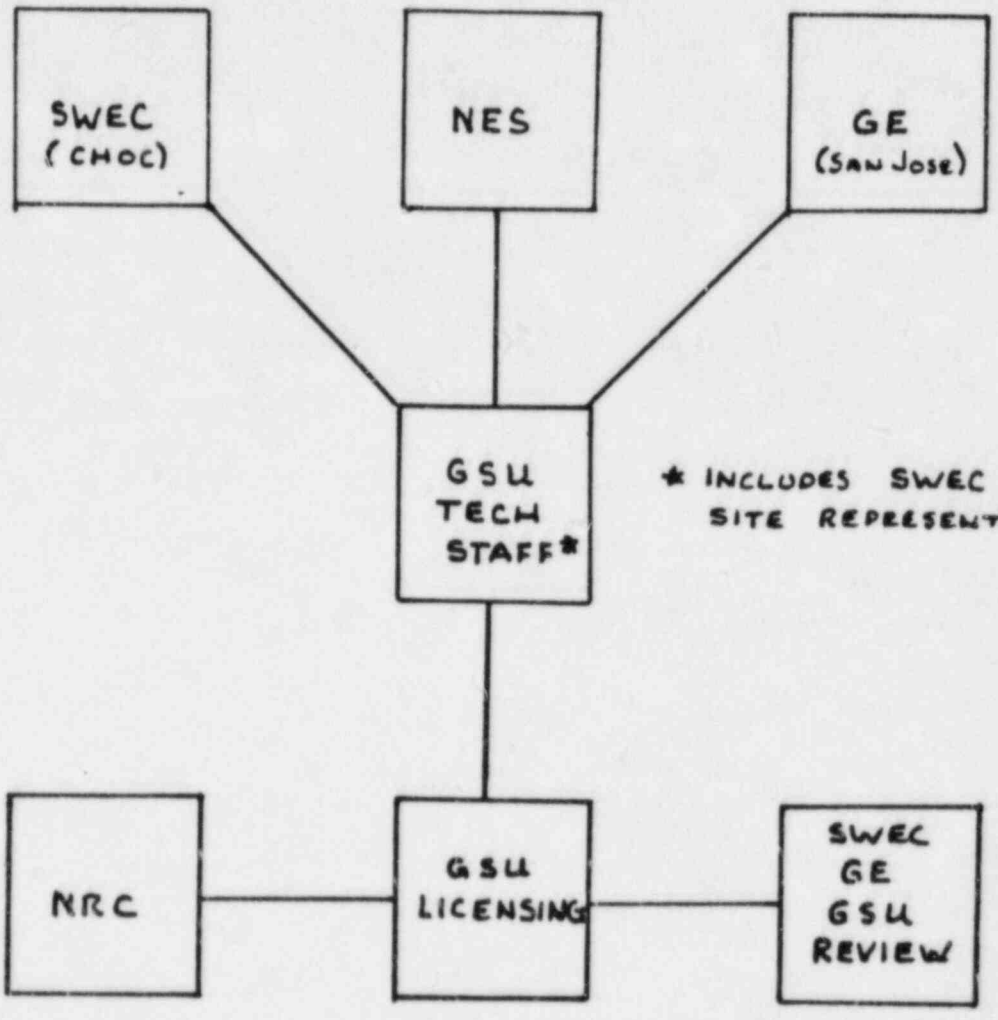
GSU

E. Grant
B. Reed
J. Price
D. Williamson
T. Overlied

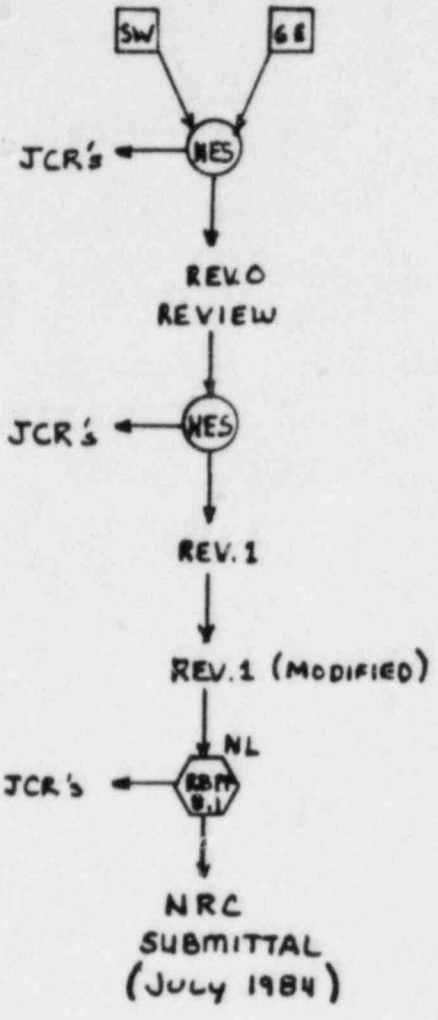
GSU Consultant

C. Alm

INTERFACE


SWEC
(CHOC)
NES
GE
(SAN JOSE)
GSU
TECH
STAFF*
* INCLUDES SWEC and GE
SITE REPRESENTATIVES
NRC
GSU
LICENSING
SWEC
GE
GSU
REVIEW

RBS TS DEVELOPMENT

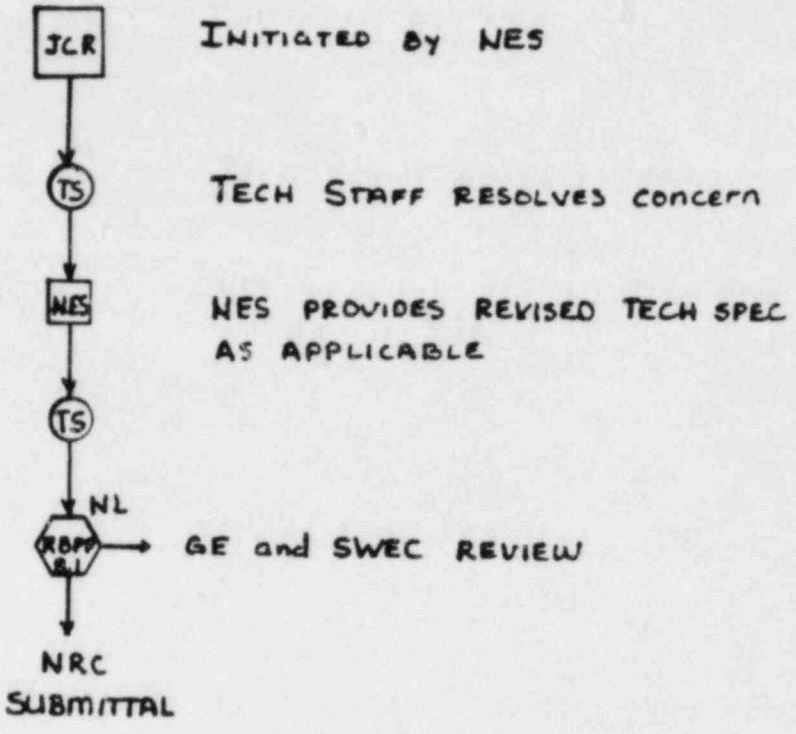


REV.0 REVIEWED BY TECH STAFF, Nuclear Licensing, SWEC/GE Site Representatives, and OTHERS

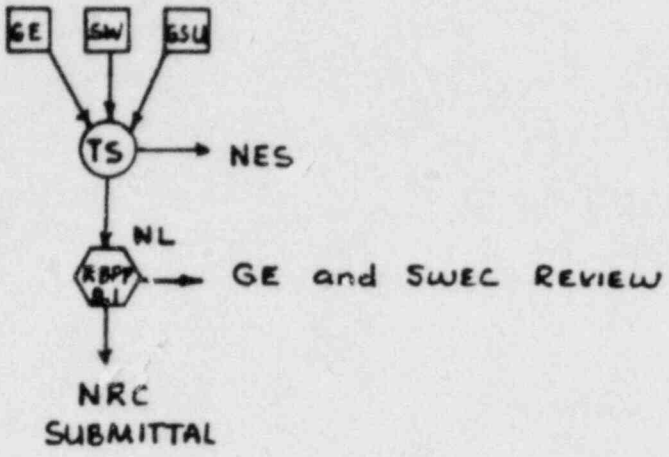
REV.1 REVIEWED BY TECH STAFF, Nuclear Licensing, SWEC/GE Site Representatives, and OTHERS IN Meetings WITH NES.

REV.1 (MODIFIED) REVIEWED IN accordance WITH RBPP B.1. GE and SWEC REVIEW.

JOB CONCERN RESOLUTION (JCR)



RBS TS CHANGES



Enclosure 4

Note:

This Enclosure Contains

Double Sided Copies

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

10-8-84

DRAFT

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

Justification (1.9) - Normal operation involvement does not effect core reactivity in any configuration.

Justification (1.11) - Wording in Summary (a) deleted per justification given in Def # 1.31 Wording is changed in Subparts (c) and (e) to be consistent with the definition of PRIMARY CONTAINMENT Subparts (c) and (e).

DRAFT

DEFINITIONS

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the highest value of the FLPD which exists in the core.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.11 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE drywell automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in ~~Table 3.6.4-1 of~~ Specification 3.6.4.

b. All drywell equipment hatches are closed and sealed.

c. The drywell airlock is ~~OPERABLE pursuant to~~ in compliance with the requirements of Specification 3.6.2.3.

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is ~~OPERABLE pursuant to~~ in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each drywell penetration; e.g., welds, bellows or O-rings, is OPERABLE.

INSERT

MEMBER(S) OF THE PUBLIC

1.24 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.26 The OFFSITE DOES CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents in the calculation of gaseous and liquid effluent monitoring alarm/trip set-points and in the conduct of the environmental radiological monitoring progam.

JUSTIFICATION (1.31) - THERE ARE EXCEPTIONS IN THE SPEC WHICH ARE NOT IN THE TABLE. REFERENCE TO THE SPEC IS SUFFICIENT TO COVER EXCEPTIONS IN THE TABLE.

Justification (1.32) - PROCESS CONTROL PROGRAM was inserted per RETS

Justification (1.) - PURGE-PURGING definition deleted because of deletion of RETS Section 3/4.11.2.8 which is specific to Mark I and Mark II containments.

Justification (1.33) - See FSAR Table 1.3-1 for Rated Thermal Power of 2894 MWT

PHYSICS TESTS

1.29 ~~1.26~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.30 ~~1.27~~ PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.31 ~~1.28~~ PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in ~~Table 3.6.4-1 of~~ Specification 3.6.4.

b. All containment equipment hatches ~~are~~ closed and sealed.

c. Each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

1.32 INSERT HERE;
RATED THERMAL POWER

1.33 ~~1.29~~ RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~(3833)~~ 2894 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 ~~1.30~~ REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. ~~$}$~~0.38~~$\%$~~ delta k/k with the highest worth rod analytically determined, or

b. ~~$}$~~0.28~~$\%$~~ delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS* and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish ~~SECONDARY CONTAINMENT INTEGRITY~~ Primary Containment — Shutdown within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, ~~including uncertainties and calculation biases~~, is equal to the specified limit.

c. Within ~~one~~ 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

*~~Except~~ movement of IRMs, SRMs or special movable detectors.

Delete "corrected" :
If the difference is "corrected" one is no longer in the ACTION for LCO.

Replace "operation may ...." with "and the provisions ... is explained" for consistancy throughout specifications.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by;

1. Verifying the continuity of the explosive charge.

2. Determining that the available weight of ~~(5500)~~ 4246 lbs and the ~~boron~~ percent weight sodium pentaborate concentration of sodium pentaborate in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

3. 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.

c. Demonstrating that , when tested ~~(at least once per 92 days)~~ pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1220 psig is met.

d. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.

2. Demonstrating that the pump relief valve setpoint is less than or equal to ~~(1600)~~ 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

3. **Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by: ~~(pumping from the storage tank to the test tank) and then draining and flushing the piping with demineralized water~~

REPLACE WITH INSERT PAGE 3/4 1-19a

4. ~~Demonstrating that the storage tank heaters are OPERABLE by verifying a temperature rise of the sodium pentaborate solution in the storage tank by at least ___°F within ___ minutes after the heaters are energized.~~

REPLACE with INSERT A PAGE 3/4 1-19a

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F ~~the limit of Figure 3.1.5-1~~.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

JUL 9 1984
AUG 15 1984

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the SHUTDOWN position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce ~~turbine first stage pressure~~ Thermal Power to less than the required automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the SHUTDOWN position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable controls and lock the reactor mode switch in the Shutdown position within one hour.

*Except movement of IRM, SRM or special moveable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

ACTION 10 - Within one hour place the inoperable instrument channels in both trip systems in the tripped condition and the provisions of Specification 3.0.4 are not applicable. *

Otherwise comply with ACTION 4 for Main Steam Isolation Valve-Closure or ACTION 6 for Turbine Stop Valve-Closure

* AN INOPERABLE CHANNEL NEED NOT BE PLACED IN THE TRIPPED CONDITION WHERE THIS WOULD CAUSE THE MAIN STEAM TO CLOSE.

8-15-84

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry ~~(or the rod pattern control system shall be OPERABLE)~~ prior to and during the time any control is withdrawn* and shutdown margin demonstrations are being performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required, per specification 3.10.1. ← added

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) This function shall be automatically bypassed when turbine first stage pressure is < (250) psig, equivalent to THERMAL POWER less than (40)% of RATED THERMAL POWER.

(i) This function shall be automatically bypassed when the mode switch is in the run position.

(j) This function is not required OPERABLE when the shorting links have been removed, the SRM(s) are OPERABLE per Specification 3.9.2 and shutdown margin demonstrations are not being performed per Specification 3.10.3.

added

** Initial setpoint. Final setpoint to be determined during startup test program. Any required change to setpoint shall be submitted to the Commission within 90 days of test completion.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤ [0.09][**]
c. Neutron Flux - High	≤ [0.09]
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ [0.35]
4. Reactor Vessel Water Level - Low, Level 3	≤ ~~(0.30)~~ 1.05
5. Reactor Vessel Water Level - High, Level 8	≤ ~~(0.30)~~ 1.05
6. Main Steam Line Isolation Valve - Closure	≤ [0.06]
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	NA
10. Turbine Stop Valve - Closure	≤ [0.06]
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤ [0.07]#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. ~~(This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.118, November 1977.)~~

**[Not] Including simulated thermal power time constant, 6 ± ~~1~~ .6 seconds.

#Measured from start of turbine control valve fast closure.

Item 2a and 2d:

Page 3/4 3-7: To item 2a and d add footnote (m) to Operational Condition 5

Page 3/4 3-8: Add footnote (m)

Justification: When performing work under the vessel during operational condition 5 the LPRM instrument string cables are removed to prevent damage to, and reduce personnel exposure (by reduction of time required to perform the required task). Removal of the shorting links will ensure adequate protection for plant personnel and the public.

Item 2b: delete footnote (h) on page 3/4 3-7 and 3/4 3-8.

Justification: This surveillance is meaningless and provides no benefit to the assurrance of safety. The instrumentation for APRM flow biased scram, functions fro the input of recirculation flow not control valve position. Also technical specification 3/4.1.2 already requires a similar surveillance since, this parameter related to jet pump operability and therefore it is acceptable to require such a surveillance in this specification.

Item 3: Footnote (j) added to make consistent with footnote (d) of TABLE 3.3.1-1

Item 8: Footnote (l) added to make consistent with footnote (f) of of TABLE 3.3.1-1

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b)	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U,S,(b)	S/U[c], W	SA	2
	S	W	SA	3, 5[m] ← added
b. Flow Biased Simulated Thermal Power - High	S,D	S/U[c], W	W[d][e], SA, R[i]	1
c. Neutron Flux - High	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5[m] ← added
3. Reactor Vessel Steam Dome Pressure - High	S	M	R[g]	1, 2[(j)] ← added
4. Reactor Vessel Water Level - Low, Level 3	S	M	R[g]	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R[g]	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2[(j)]
8. Drywell Pressure - High	NA	M	R[g]	1, 2[(l)] ← added

Item 9: To make consistent with River Bend plant specific design

Item 10 and 11: Delete the footnote (g) since these instruments do not have the GE trip unit configuration.

Footnote (g): Deleted phrase in parenthesis since this identifies when footnote g is applicable during technical specification mark up.

Footnote (i): The design of RBS uses hard wired resistors and capacitor to develop the time constant and these components do not have range adjustment capabilities. LESS THAN 6 SECONDS IS CONSERVATIVE. THIS CHANGE WILL REFLECT THE TIME CONSTANT TO BE OUT OF SERVICE WITHOUT DECLARING APRM INOPERABLE

Footnote (h): Refer to item 2b of page 3/4 3-7

Footnote (l): Refer to item 8 page 3/4 3-7

Footnote (m): Refer to item 2a and 2d of page 3/4 3-7

Footnote (n): With bypass valves open the first stage turbine pressure is not

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High a. Trans/Trip Units	S	M	R(g)	1, 2, 5(k) } revised
b. Float Switch	NA	Q	R	1, 2, 5(k) }
10. Turbine Stop Valve - Closure	~~NA~~ S(m)	M	R	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA S(m)	M	R	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least ~~1~~1/2~~)~~ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ~~1~~1/2~~)~~ decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system. setpoints
(g) Calibrate trip unit at least once per 31 days. ~~(BWR/6 relay only)~~
~~(h) Verify measured core flow to be greater than or equal to established core flow at the existing flow control valve position.~~
(i) This calibration shall consist of ~~verifying~~ ~~(adjustment, as required of)~~ the ~~equivalent~~ simulated thermal power time constant to be less than 6.6 seconds
(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(k) With any control rod with drawn. Not applicable to ~~control~~ rods removed per Specification 3.9.10.1 or 3.9.10.2.

deleted

(l) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required per specification 3.10.1.
(m) (SAME AS) ON PAGE 3/4 3-5)
(n) VERIFY THE TURBINE BYPASS VALVES ARE CLOSED WHEN THERMAL POWER IS GREATER THAN OR EQUAL TO 40% RATED THERMAL POWER

3/4 3-8

8-15-84

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL #	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low-Low, Level 2	1,7,8,9 ~~(3)~~ (b)(~~c~~)(k)	2	1, 2, 3 ~~and *~~	20
b. Drywell Pressure - High	1,3,8 ~~(2, 6)~~ (b)(c)(k)	2	1, 2, 3	20
c. ~~Plant Exhaust Plenum~~ Containment Purge Isolation Radiation - High	8 ~~(6)~~ (b)(c)	1	1, 2, 3 and ~~*~~ #	21
d. Manual Initiation	1,3,7,8 ~~(2, 3, 6)~~	(2)/(group)	1, 2, 3 ~~and *~~	(22)
~~e.~~	~~———~~	~~———~~	~~———~~	~~——~~
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level-Low Low Low, Level 1	6 ~~(1, 5)~~ (c)	2	1, 2, 3	20
b. Main Steam Line Radiation - High	6,9 ~~(1, 7)~~ (d)	~~2 */(1line)~~	1, 2, 3	23
c. Main Steam Line Pressure - Low	6 ~~(1)~~	~~2 */(1line)~~	1	24
d. Main Steam Line Flow - High	6 ~~(1)~~	2/(1line) (c)	1, 2, 3	23
e. Condenser Vacuum - Low	6 ~~(1)~~	2	1, 2, ** 3**	23
f. Main Steam Line Tunnel Temperature - High	6 ~~(1)~~	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	6 ~~(1)~~	2	1, 2, 3	23
h. Manual Initiation	6 ~~(1, 5, 7)~~	(2)/(group)	1, 2, 3	(22)
i. Main Steam Leak Area temperature - high (Turbine Building)	~~6~~	2	1, 2, 3	23)

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL *	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	11,12,13 (6) (b)(c)(f)(i)(j)	2	1, 2, 3, ~~and *~~	25
b. Drywell Pressure - High	11,12,13 (6) (b)(c)(f)(i)(j)	2	1, 2, 3	25
c. Fuel ~~Handling Area~~ Building Ventilation Exhaust Radiation - High ~~High~~	13 (6) (b)(f)(i)	1	1, 2, 3, ~~and *~~	26 28
d. ~~Fuel Handling Area Pool Sweep~~ Reactor Building Annulus Ventilation Exhaust Radiation - High ~~High~~	12 (6) (b)(f)(j)	1	1, 2, 3, ~~and *~~	26 29
(e.f) Manual Initiation	11,12,13 (6) (d) ~~(6) (d)~~	(1)/(group) ~~(1)/(group)~~	1, 2, 3	(26) ~~(26)~~ (29)
(e.g) REACTOR BUILDING ANNULUS PRESSURE CONTROL SYSTEM AIR FLOW LOW	12 (b)(f)(j)	1	1 2 3	
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	7 (3)	(1)	1, 2, 3	27
b. Δ Flow Timer	7 (3)	1	1, 2, 3	27
c. Equipment Area Temperature - High	7 (3)	(1)	1, 2, 3	27
d. Equipment Area Δ Temp. - High	7 (3)	(1)	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	7 (3)	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	7 (3)	(1)	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	7 (3) (g)	(1) ~~NA 1 (9)~~	1, 2, 3	27
h. SLCS Initiation	7 (3)	1	1, 2, 3	27
i. Manual Initiation	7 (3)	~~(1)~~/(group)	1, 2, 3	(26)
j.				

Item 5.c footnote (h): Deleted, not applicable.

Item 5.n

footnote (i): Deleted since the markup of Minimum Operable Channels per trip system identifies as each valve's control switch. Therefore, the footnote is unnecessary.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:
a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In Operational Condition ~~*~~, suspend primary CORE ALTERATIONS, handling of irradiated fuel in the containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

~~ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.~~

ACTION 28 & 29 - (SEE PAGE 3/4 3-15a) (REPLACE WITH INSERT B 3/4 3-15a)

NOTES

~~* When handling irradiated fuel in the containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.~~

** May be bypassed with ~~reactor steam pressure < (1043) psig and~~ all turbine stop valves ~~closed~~, ~~less than~~ 90% open and Reactor Mode Switch not in RUN.

During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the standby gas treatment system.

(c) Also actuates the control room ~~emergency filtration~~ air conditioning system in the ~~isolation~~ emergency mode of operation.

(d) Also trips and isolates the ~~mechanical vacuum~~ air removal pumps.

~~(e) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.~~

(f) Also actuates secondary containment ventilation isolation dampers ~~and valves~~ per Table 3.6.~~4~~.2-1.

(g) ~~Closes only RWCU system isolation valve(s)~~ (REPLACE WITH INSERT A 3/4 3-15a)

~~(h)~~ Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.

~~((i) Manual initiation isolates only and only with a coincident reactor vessel water level-low, level 3).~~

(i) Also Starts the Fuel Building Exhaust Filter Trains A and B

(j) Also starts the Annulus Mixing System

(k) Also actuates the containment hydrogen analyzer/monitor recorder.

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - {Low Low, Level 2}	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Water Level-High, Level {8}	{S}	M	{R}(a)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R(a)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(a)	1, 2, 3, 4*, 5*
(f. Pump Discharge Pressure-High	{S}	M	{R}	1, 2, 3, 4*, 5*)
(g. HPCS System Flow Rate-Low	{S}	M	{R}	1, 2, 3, 4*, 5*)
~~(h. HPCS Bus Power Monitor~~	NA	~~M~~	(NA)	~~1, 2, 3, 4*, 5*)~~
i. Manual Initiation	NA	{R}	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. DIVISION 1 and 2				
a. 4.16 kv ~~Emergency~~ Standby Bus Under-voltage ~~(Loss of Voltage)~~ (Sustained Undervoltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kv ~~Emergency~~ Standby Bus Under-voltage (Degraded Voltage)	S NA	M NA	R	1, 2, 3, 4**, 5**
2. DIVISION 3				
~~a. 4.16 Kv Standby Bus Undervoltage~~ LATER	NA	NA	R	1, 2, 3, 4**, 5**

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to {100} psig.

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required when ESF equipment is required to be OPERABLE.

(a) Calibrate trip unit at least once per 31 days.

~~((b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.)~~

SETPOINT

RIVER BEND-UNIT 1

3/4 3-37

8-24-84

Action b and c: The River Bend Station does have two independant trip systems however the logic is such that placing any instrument channel in the tripped condition results in loss of both Reactor Recirculation Pumps. Therefore change Action b such that the trip system is declared inoperable and delete Action c.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	M	R*
2. Reactor Vessel Pressure - High	S	M	R*

~~*Calibrate trip unit at least once per 31 days.~~

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - (Low Low, Level 2)	S	M	R(a)
b. Reactor Vessel Water Level - High, Level 8	S	M	R
c. Condensate Storage Tank Level - Low	S	M	R
d. Suppression Pool Water Level - High	S	M	R
e. Manual Initiation	NA	~~M(b)~~ ~~(R)~~	NA

(a) Calibrate trip unit SETPOINT at least once per 31 days.

~~((b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.)~~

RIVER BEND-UNIT 1

~~GE-ETS-(08/84)~~

3/4 3-53 52

8-24-84

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD PATTERN CONTROL SYSTEM			
a. Low Power Setpoint	2	1, 2	60
b. ~~Intermediate Rod Withdrawal Limiter Setpoint~~ High Power Setpoint	2	1, X	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	6	1	61
b. Inoperative	6	1, 2, 5(e)	61
c. Downscale	6	1	61
d. Neutron Flux - Upscale, Startup	6	2, 5(e)	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(a)	4	2, 5	61
b. Upscale(b)	4	2, 5	61
c. Inoperative(b)	4	2, 5	61
d. Downscale(c)	4	2, 5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in(d)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(d)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	(2)	1, 2, 5*	62
b. Scram Trip Bypass	(2)	(1, 2,) 5*	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2(5)	1	62 61
~~b. Inoperative~~	~~2~~	~~1~~	~~62~~
~~c. (Comparator) (Downscale)~~	~~2~~	~~1~~	~~62~~

Replace with insert shown on page 3/4 3-54a

Item 3 : River Bend has a total of 4 SRM's. One is permitted to be bypassed in OC 2 and Tech Spec 3.9.2 requires only 2 to be operating in OC 5

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	~~(20)~~% of RATED THERMAL POWER (27.5 ± 3)	27.5 ± 7.5 ~~(20 + 15, - 0)~~% of RATED THERMAL POWER
b. ~~Intermediate Rod Withdrawal Limiter~~ High Power Setpoint	~~(70)~~% of RATED THERMAL POWER (62.5 ± 3)	~~(70)~~% of RATED THERMAL POWER 62.5 ± 7.5
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + ~~(42)~~%*	≤ 0.66 W + ~~(45)~~%*
b. Inoperative	NA	NA
c. Downscale	≥ ~~(5)~~% of RATED THERMAL POWER	≥ ~~(3)~~% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ ~~(12)~~% of RATED THERMAL POWER	≤ ~~(14)~~% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ ~~(2)~~ x 10^5 cps	≤ ~~(5)~~ 1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ ~~(3)~~ cps	≥ ~~(3)~~ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ ~~(108/125)~~ division of full scale	≤ ~~(110/125)~~ division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ ~~(5/125)~~ division of full scale	≥ ~~(3/125)~~ division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ ~~(32.6)~~ inches	≤ ~~(34)~~ inches
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ ~~(108)~~% of rated flow	≤ ~~(111)~~% of rated flow
~~b. Inoperative~~	~~NA~~	~~NA~~
~~c. (Comparator) (Downscale)~~	~~≤ (10)% flow deviation~~	~~≤ (11)% flow deviation~~

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

refer to TABLE 3.3.6-1 Explanations.

Item 1a and 1b: Addition of $S^{(9)}$ due to requested deletion of Technical Specification 3/4.1.4.1.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.10 ~~3.3.3.10~~ The radioactive liquid effluent monitoring instrumentation channels 3.3.7.10-1 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 suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table ~~3.3-12~~. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, ~~in lieu of a Licensee Event Report,~~ explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified. 3.3.7.10-1

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 ~~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 CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table ~~4.3-12~~ 4.3.7.10-1

Item 1: Deleted "Gross" in that it is a known editorial change being made. Also comment received from GSU supports this change.

Item 2: Deleted "Gross Beta or Gamma" in that it is a known editorial change to occur.

Item 2a: Title change

Item 2b: Deleted since the River Bend Station does not have any other effluent line other than that identified in 2a which permits direct discharge. FSAR Section 11.2.

Items 4 and 5 deleted: River Bend does not have commitment as having in Licensing document FSAR Sction 11.2 and 11.5.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.11 ~~3.3.3.11~~ The radioactive gaseous effluent monitoring instrumentation channels shown in Table ~~3.3-13~~ 3.3.7.11-1 shall be OPERABLE with their alarm/trip* setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table ~~3.3-13~~ 3.3.7.11-1

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint* less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table ~~3.3-13~~ 3.3.7.11-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, ~~in lieu of a Licensee Event Report,~~ explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.11 ~~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 CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table ~~4.3-13~~ 4.3.7.11-1

---

* Alarm/trip setpoint for the hydrogen analyzers are not required for OPERABILITY.

10-8-84

Tech Spec & Table numbers changed to agree with sequence in RB-U1 Tech Specs and GE-STS Numbering system.
Table 3.3.7.11-1 & 4.3.7.11-1 - Items 1.b,c,e; 3.b,c,d,e; 5 (all); 6 (all); and 9 (all) - RB-U1 design combines these vents and exhausts for monitoring at final release point, plant exhaust duct. Deleted upstream monitors per SRP 11.5 and NUREG-0133.

TABLE 4.3.~~7~~.1-1 ~~(Continued)~~

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT VENTILATION ~~SPRAY~~ SYSTEM UNITCOOLER A				
a. Drywell Pressure-High ~~/-to-Annulus~~ ΔP	~~(NA)~~ S	~~(M)~~	~~(Q)~~ R	1, 2, 3
b. Containment ~~Pressure~~-High	~~(NA)~~ S	~~(M)~~	~~(Q)~~ SA	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low, Level 1	~~(NA)~~ S	~~(M)~~	~~(Q)~~ R	1, 2, 3
d. Timer~~s~~	~~(NA)~~	~~(M)~~	~~(R)~~ Q	1, 2, 3
~~2.~~3 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level ~~(8)~~	~~(NA)~~ S	~~(M)~~	~~(R)~~	1
2. PRIMARY CONTAINMENT VENTILATION ~~SPRAY~~ SYSTEM UNITCOOLER B				
a. Drywell Pressure-High ~~/-to-Annulus~~ ΔP	S	M	R	1, 2, 3
b. Containment ~~Pressure~~-High	S	M	SA	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low, Level 1	S	M	R	1, 2, 3
d. Timer	NA	M	Q	1, 2, 3

DRAFT

LCO 3.4.1.1: addition of the phrase "with the associated flow control valve OPERABLE" to clarify that failure to meet surveillance requirements would not meet the LCO

FSAR section 14.2.12.3.27.1, 14.2.12.3.27.2, 15.3.2.1.1, 15.3.2.3.2.2, 15.4.5.3.2 and 15.4.5.3.3.2

Justification: For clarification since, in the nuclear industry there exists a conflict on whether the recirculation pumps can be restarted.

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries per gram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or for more than 800 hours cumulative operating time in a consecutive 12-month period, or greater than 4.0 microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. Greater than 100/$\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

b. ~~In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram~~, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

~~c. In OPERATIONAL CONDITION 1 or 2, w/th:~~

~~1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour*, or~~

~~2. The off-gas level, at the SJAE, increased by more than (10,000) microcuries per second in one hour during steady state operation at release rates less than (75,000) microcuries per second, or~~

~~3. The off-gas level, at the SJAE, increased by more than (15)% in one hour during steady state operation at release rates greater than (75,000) microcuries per second.~~

~~perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.2 at least once per 92 days containing the results of the specific activity analysis together with the below additional information for each occurrence.~~

Additional Information

1. Reactor power history starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.
2. Fuel burnup by core region.
3. Clean-up flow history starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.
4. Off-gas level starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

*Not applicable during the startup test program.

Item 1) As written implies separate determination of Beta and Gamma activity

Item 4) for clarity

Item 4b) DELETED because Action C DELETED - SEE JUSTIFICATION FOR ACTION C.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta/~~and~~ Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for E Determination	At least once per 6 months*	1
4. Isotopic Analysis for ~~Iodine~~ I-131, I-132, I-133, I-134, and I-135	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	~~b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.~~	~~1, 2~~
5. Isotopic Analysis of an Off-gas Sample ~~Including Quantitative Measurements~~ for at least Xe-133, Xe-135 and Kr-88 concentrations	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the primary coolant system is restored to within its limits.

Note - This technical specification may be revised due to recent changes in Appendix G & H

LCO 3.4.6.1 a and b: FSAR section 5.3.3.6 page 5.3-21 FSAR states as cooldown also.

LCO 3.4.6.1c: Requested clarification by GSU since curves are normally in horizontal direction and to state above is just confusing. To clarify delete the phrase (operations above the heatup and cooldown limit curves). This does not change the intent

LCO 3.4.6.1d: acceptance of 70°F per FSAR section

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.*

4.5.2.2 The HPCS system shall be determined OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.e.

* ONE OF THE ABOVE REQUIRED LPCI SUBSYSTEMS MAY BE ALIGNED FOR ANOTHER MODE OF OPERATION PROVIDED IT CAN BE MANUALLY REALIGNED.

3/4.5.3 SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least ~~(136, 146)~~ 137,571 ft³, equivalent to a level of ~~(19'5")~~. 19'6"

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least ~~(93, 600)~~ (115,879) ft³, equivalent to a level of ~~(13'0")~~, except that the suppression pool level may be less than the limit or may be drained provided that: (16'2")

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least ~~(150,000)~~ 125,000 available gallons of water, equivalent to a level of ~~( )%~~ 33.2%, and

4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression pool water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression pool water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish ~~SECONDARY CONTAINMENT INTEGRITY~~ Primary Containment — Shutdown within 8 hours.

~~*~~ The suppression pool is not required to be OPERABLE provided that the reactor vessel head is removed, ~~the cavity is flooded (or being flooded from the suppression pool), the upper containment fuel pool gates are removed, the spent fuel pool gates are removed (when the cavity is flooded), and~~ the water level is maintained within the limits of Specification 3.9.8 and ~~3.9.9.~~ the upper containment pool gate between the fuel pool and reactor vessel pool is open

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, ~~(15.0)~~ psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than ~~or equal to~~ 0.60 La. 6.31

b. At least once per 31 days by verifying that all containment 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 position, except as provided ~~in Table 3.6.4-1 of~~ Specification 3.6.4.

c. By verifying each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1

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

Description of Change:

Added Tech Spec 3/4.6.1.1.2.

Justification:

RB-U1 safety analysis for fuel handling accident in primary containment assumes holdup in primary containment.

Other Sections Affected:

3/4.6.1.1 becomes 3/4.6.1.1.1.3/4.6.6, secondary containment series of Tech Specs becomes only applicable to handling of irradiated fuel in Fuel Building.

Title changed to be more distinctive.

SR 4.6.1.1.2.a - added "within 24 hours prior to" to ensure proper conditions are set at start of applicable period. Added "during operational condition*" for clarity. Added "hatches" to cover all penetration types since deleted SR 4.6.1.1.2(old)a.

~~3/4.6~~ CONTAINMENT SYSTEMS

~~3/4.6.1 PRIMARY CONTAINMENT~~

PRIMARY CONTAINMENT ~~INTEGRITY~~ - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.6.1.1 ~~PRIMARY CONTAINMENT INTEGRITY~~ PRIMARY CONTAINMENT CLOSURE-SHUTDOWN shall be maintained.

APPLICABILITY: ~~OPERATIONAL CONDITIONS 1, 2 and 3.~~ OPERATIONAL CONDITION *.

ACTION:

Without ~~PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~ PRIMARY CONTAINMENT — SHUTDOWN, SUSPEND HANDLING OF IRRADIATED FUEL IN THE PRIMARY CONTAINMENT, CORE ALTERATIONS, AND OPERATIONS WITH A POTENTIAL FOR DRAINING THE REACTOR VESSEL.

SURVEILLANCE REQUIREMENTS

4.6.1.1 ~~PRIMARY CONTAINMENT INTEGRITY~~ PRIMARY CONTAINMENT — SHUTDOWN shall be demonstrated:

~~a. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, (15.0) psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.~~

b. At least once per 31 days by verifying that all containment 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 position, except as provided in ~~Table 3.6.4-1 of~~ Specification 3.6.4. WITHIN 24 HOURS PRIOR TO AND DURING OPERATIONAL CONDITION * FUEL HANDLING HATCHES, SHOWN IN TABLE 3.6.4-1 WHICH ARE CLOSED.

~~d. By verifying each containment air lock is in compliance with the requirements of Specification 3.6.1.3.~~ By verifying AT LEAST ONE DOOR IN each containment air lock is

~~e. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.~~

* WHEN HANDLING IRRADIATED FUEL IN THE PRIMARY CONTAINMENT AND DURING CORE ALTERATIONS AND OPERATIONS WITH A POTENTIAL FOR DRAINING THE REACTOR VESSEL.

~~* See Special Test Exception 3.10.1~~

~~Except valves, blind flanges, and deactivated automatic valves which are located inside the containment or drywell, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed (when the containment has not been de-inerted since the last verification or) more often than once per 92 days.~~

LCO 3.6.1.2:

a.1 - L_a - FSAR 6.2.1.1.2, page 6.2-6; 15.6.5.5.2, page 15.6-8,9; table 6.2-1; 6.2.6.1, page 6.2-89: SER 6.2.1, page 6-4; 6.2.6.1, page 6-47, show 0.26%.

P_a - FSAR 6.2.6.1, page 6.2-89; 6.2.6.2, page 6.2-91; 6.2.6.3, page 6.2-92; 6.2.1.1.1 (1.a), page 6.2-2; table 6.2-1. SER (DRAFT) 6.2.1, Page 6-4.

a.2 - Deleted.

b - Deleted "or equal to" for consistency with 10CFR50, App. J.

- Deleted items in parentheses to reorganize specification to be RB-U1 specific.
- Deleted items in parentheses to reorganize requirement to make specific to RB-U1.
- MSIV leakrate limit is controlled by MS-PLCS. NRC has agreed to not requiring MSIV leakage to be part of 0.60 L_a limit and to exempt MSIVs from leakage rate testing. NRC has also agreed to include Leakage of valves served by PVLCS in 0.60 L_a limit only until time after accident when PVLCS is assumed to be operating, 20 minutes, because operator action is required to start operation of MS-PLCS for MSIVs, but, unlike PVLCS served valves, trapped pressurized steam is assumed to prevent MSIV leakage from time of MSIV closure until start of MS-PLCS operation.
  NRC has agreed that leakage testing of MSIVs is not required.

NRC has agreed that ECCS & RCIC system isolation valves in lines which penetrate the primary containment are considered to be sealed with a fluid from a seal system adequate to maintain system pressure because of line submergence in the supression pool.

c - FSAR table 9.3-3 shows individual valve leakrate limits.

Reference to (0.3) L_a - FSAR 9.3.3. References to secondary containment bypass leakage paths - FSAR 6.2.3.2.1, Page 6.2-58; 6.2.3.2.4.5, Page 6.2-60; 7.3.1.1.3, Page 7.3-17; 9.3.6.1.2, Page 9.3-31; 9.3.6.2.2, Page 9.4-34; 9.3.6.2.3, Page 9.4-35.

Applicability - For consistency with number change for referenced Tech Spec.

Footnote* - Change to LCO 3.6.1.2 removed need for footnote.

Actions a,b,c,d & e are associated with LCO 3.6.1.2 a,b,c,d & e, respectively, and the above explanations apply. Made actions consistent with 10CFR Appendix J as in LCO 3.6.1.2.b, above.

a,b - Added "equaling or" for consistency with 10CFR50, App. J.

POSITIVE MSIV LEAKAGE CONTROL SYSTEM ~~(Optional)~~

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent main steam positive leakage control system (MSPLCS) ~~subsystems~~ DIVISION'S shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSPLCS ~~subsystem~~ DIVISION inoperable, restore the inoperable ~~subsystem~~ DIVISION to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSPLCS ~~subsystem~~ DIVISION shall be demonstrated OPERABLE:

~~a. At least once per 31 days by verifying:~~

~~1. Blower OPERABILITY by starting the injection system from the control room and operating the system for at least (15) minutes through the drain line.~~

~~2. The (functional availability) of the MSPLCS by _____.~~

~~3. _____~~

c, ~~b~~. During each COLD SHUTDOWN ~~/~~, if not performed within the previous 92 days, by cycling each remote, manual and automatic motor operated valve through at least one complete cycle of full travel ~~(in accordance with Specification 4.0.5)~~.

d, ~~f~~. At least once per 18 months by performance of a functional test which includes simulated actuation of the ~~subsystem~~ DIVISION throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that greater than or equal to ~~(8.5)~~ psig sealing pressure is established in each steam line.

e, ~~d~~. By verifying the ~~(flow~~ AND pressure, ~~temperature and level)~~ ~~(operating)~~ instrumentation to be OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
2. CHANNEL CALIBRATION at least once per 18 months.

a. BY VERIFYING THAT THE ASSOCIATED PENETRATION VALVE LEAKAGE CONTROL SYSTEM ACCUMULATOR IS OPERABLE AT LEAST ONCE PER 24 HOURS PER SPECIFICATION 4.6.1.9.a.

b. BY VERIFYING THAT THE ASSOCIATED PENETRATION VALVE LEAKAGE CONTROL SYSTEM COMPRESSOR IS OPERABLE AT LEAST ONCE PER 92 DAYS PER SPECIFICATION 4.6.1.9.b.

f. AT LEAST ONCE PER 18 MONTHS BY VERIFYING A TOTAL SEALING AIR LEAKAGE RATE OF 131 SCFH for each division into THE PRIMARY CONTAINMENT AT A TEST PRESSURE OF 11.5 PSID FOR MS-PLCS.

4.6.1.5.1 - RB-01 steel shell containment does not have a liner plate.

4.6.1.5.2 - Tolerances on cracking are applicable only to concrete containment.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the PRIMARY containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the PRIMARY containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the PRIMARY containment, ~~including the liner plate~~, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those 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.

4.6.1.5.2 Reports Any abnormal degradation of the PRIMARY containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the ~~concrete~~ PRIMARY CONTAINMENT STRUCTURE, the inspection procedure, ~~the tolerances on cracking~~, and the corrective actions taken.

SURVEILLANCE REQUIREMENTS (Continued)

~~(4.6.1.8.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed (20) inch (drywell and) containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.05) L_a when pressurized to P_a.)~~

4.6.1.8.~~3~~2 At least once per 92 days each ~~(6)~~ 24 INCH AND 36 inch ~~(~~drywell and~~)~~ PRIMARY containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.01) L_a when pressurized to P_a.~~)~~

4.6.1.8.3 Once per 24 months verify the other test [illegible] are satisfied

Description of Change:

Deleted Tech Spec

Justification:

Not in RB-U1 design. RB-U1 design uses air positive seal isolation valve leakage control system.

INSERT A

DOOR, IF OPENED FOLLOWING THE DRYWELL BYPASS LEAKAGE RATE TEST, BY VERIFYING THAT THE COMBINATION EQUIPMENT HATCH/ PERSONNEL DOOR IS IN PLACE AND BY LEAK RATE TESTING THE GAP BETWEEN THE SEALS AND VERIFYING THAT THE MEASURED LEAKAGE RATE FOR THESE SEALS, WHEN PRESSURIZED WITH GAS AT 3.0 PSIG:

1. IF THE PERSONNEL DOOR WAS OPENED, IS LESS THAN OR EQUAL TO 200 CC PER HOUR, AND
2. IF THE EQUIPMENT HATCH WAS OPENED, IS LESS THAN OR EQUAL TO 75 CC PER HOUR.

INSERT B

e. By verifying the ~~PERSONNEL~~ door inflatable seal system OPERABLE by:

1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to (65) psig.

2. At least once per 18 months conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than (2) psig from (104) psig within (48) hours.

INSERTS FOR 3/4 6-15

8-15-84

LCO 3.6.2.2

FSAR 6.2.1.1.3.4, page 6.2-31. Question 480.16, Q&R page 6.2-15, and SER 6.2.1.7, page 6-14

SR 4.6.2.2

FSAR 6.2.1.1.3.4, page 6.2-32, 32a; 6.2.6.5.1, pages 6.2-93, 94; table 6.2-1.

GE-STS requires test at least once per 18 months. FSAR 6.2.1.1.3.4, p. 6.2-32a, shows 3 times per 10 years.

DRYWELL BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.2.2 Drywell bypass leakage shall be less than or equal to 10% of the minimum acceptable $A/\sqrt{k}$ design value of 1.0 $ft.^2$

APPLICABILITY: When DRYWELL INTEGRITY is required per Specification 3.6.2.1.

ACTION:

With the drywell bypass leakage greater than 10% of the minimum acceptable $A/\sqrt{k}$ design value of 1.0 $ft.^2$, restore the drywell bypass leakage to within the limit prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.~~6.2.2~~ The drywell bypass leakage rate test shall be conducted at least once per ~~18~~ 24 months at an initial differential pressure of 3.0 psi and the $A/\sqrt{k}$ shall be calculated from the measured leakage. One drywell airlock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other leakage rate test.

a. If any drywell bypass leakage test fails to meet the specified limit, the schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the limit, a test shall be performed at least every 9 months until two consecutive tests meet the limit, at which time the 18 month test schedule may be resumed.

b. The provisions of Specification 4.0.2 are not applicable.

NOTE: Changes throughout because there is only one drywell airlock.

LCO 3.6.2.3

b - FSAR 6.2.1.1.3.4, pages 6.2-32, 32a; 6.2.6.5.1, pages 6.2-93, 94; table 6.2-1.

CONTAINMENT SYSTEMS

DRYWELL AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.2.3 ~~Each~~ THE drywell 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 drywell, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to ~~(2) scf~~ ( ) per hour at ~~(Pa), (25.0) psig.~~ A DRYWELL INTERNAL PRESSURE DIFFERENTIAL OF 3 PSID.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one drywell 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 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 SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the drywell air lock inoperable, except as a 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 SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~(c. With one drywell air lock door inflatable seal system air flask pressure instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify air flask pressure to be ≥ (90) psig at least once per 12 hours.)~~

*See Special Test Exception 3.10.1.

SR 4.6.2.3

a,b - FSAR 6.2.1.1.3.4, pages 6.2-32, 32a; 6.2.6.5.1, pages 6.2-93, 94; table 6.2-1.

b.1 and b.2 - 10CFR50 Appendix J is not applicable to drywell.

c - Applicability of this specification is changed 1.2.4 The drywell [illegible] will normally not be used in these operational conditions. The surveillance frequency was more appropriate to the months to allow this test to be performed during [illegible] conditions at a refueling outage [illegible] NUREG recommendations.

SURVEILLANCE REQUIREMENTS

4.6.2.3 ~~Each~~ THE drywell air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air ( ) lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to ~~(2) scf~~ per hour when the gap between the door seals is pressurized to ~~Pa, (15.0)~~ 3.0 psig.

b. By conducting an overall air lock leakage test at ~~(P a), (15.0) psig~~ A DRYWELL INTERNAL PRESSURE DIFFERENTIAL OF 3.0 PSID and verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months.

2. Prior to establishing DRYWELL INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.

c. At least once per ~~6~~ 18 months by verifying that only one door in THE ~~each~~ air lock can be opened at a time. ~~(##)~~

d. By verifying the door inflatable seal system OPERABLE by:

~~1. Demonstrating two seal air flask pressure instrumentation channels OPERABLE with a low pressure setpoint of ≥(90) psig by performance of a:~~

~~a) CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~b) CHANNEL CALIBRATION at least once per 18 months.~~

1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to ~~90~~ 80 psig.

2. At least once per 18 months conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than ~~(2)~~ (3) psig from ~~(104)~~ (110) psig within ~~(48)~~ hours.

~~# The provisions of Specification 4.0.2 are not applicable.~~

~~* Exemption to Appendix J of 10 CFR 50.~~

~~(### Except that the inner door need not be opened to verify interlock OPERABILITY when the drywell is inerted, provided that the inner door interlock is tested within 8 hours after the drywell is de-inerted.)~~

SR 4.6.2.4.1

FSAR 6.2.6.5.1, page 6.2-93.

CONTAINMENT SYSTEMS

DRYWELL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.4 The structural integrity of the drywell shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.2.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the drywell not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the ~~Type A containment~~ DRYWELL BYPASS drywell shall be determined during the shutdown for each leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the ~~Type A containment~~ DRYWELL BYPASS leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.2.4.2 Reports Any abnormal degradation of the drywell structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. 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

3/4.6.3 DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.6.3.1 The suppression pool shall be OPERABLE with the pool water:

a. Volume between ~~(130,146)~~ 137,576 ft^3 and ~~(138,851)~~ 141,836 ft^3, equivalent to a level between ~~(18' 8")~~ 19'6" and ~~(19' 6")~~ 20'0", and a

b. Maximum average temperature of ~~(95)~~ °F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

1. ~~(105)~~ °F during testing which adds heat to the suppression pool.
2. ~~(110)~~ °F with THERMAL POWER less than or equal to ~~(1)~~% of RATED THERMAL POWER.
3. ~~(120)~~ °F with the main steam line isolation valves closed following a scram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression pool water level outside the above limits restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression pool average water temperature greater than ~~(95)~~ °F, restore the average temperature to less than or equal to ~~(95)~~ °F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression pool average water temperature greater than ~~(105)~~ °F during testing which adds heat (OTHER THAN PUMP HEAT) to the suppression pool, stop all testing which adds heat to the suppression pool and restore the average temperature to less than ~~(95)~~ °F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression pool average water temperature greater than:

a) ~~(95)~~ °F for more than 24 hours and THERMAL POWER greater than ~~(1)~~% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b) ~~(110)~~ °F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression pool average water temperature greater than ~~(120)~~ °F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c - RB-H1 design has two temperature sensors in each of five sectors. Each sensor has two RTD elements. Only one RTD element is required to be operable to consider the sensor OPERABLE.

3/4.6.4 PRIMARY CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 The PRIMARY containment and drywell isolation valves ~~and the reactor instrumentation line excess flow check valves~~ shown in Table 3.6.4-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.4-1.

APPLICABILITY: As shown in Table 3.6.4-1. ~~(OPERATIONAL CONDITIONS 1, 2 and 3 (and **))~~

ACTION:

With one or more of the PRIMARY containment or drywell isolation valves shown in Table 3.6.4-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either: (, the provisions of 3.0.4 are not applicable,

a. Restore the inoperable valve(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

~~Otherwise,~~ in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~Otherwise,~~ in Operational Condition **, suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in the ~~secondary~~ PRIMARY containment and with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

~~b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.4-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either:~~

~~1. The inoperable valve is returned to OPERABLE status, or~~

~~2. The instrument line is isolated and the associated instrument is declared inoperable.~~

~~Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

**When handling irradiated fuel in the ~~secondary~~ PRIMARY containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SR 4.6.4.5 - Not applicable to RB-U1. System entirely within primary containment.

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
1. Primary Containment						
1B21*AOVF022A	(b)(g)	1KJB*Z1A	1,2,3 AND ##	5	No	6.31
B	(b)(g)	B		5		
C	(b)(g)	C		5		
D	(b)(g)	D		5		
1B21*AOVF028A	(g)	1KJB*Z1A		5		
B	(g)	B		5		
C	(g)	C		5		
D	(g)	D		5		
1B21*MOVF067A	(d)	1KJB*Z1A		7.5		
B	(d)	B		7.5		
C	(d)	C		7.5		
D	(d)	D		7.5		
1B21*MOVF016	(b)(g)	1KJB*Z2		16.5		
F019	(g)	"		17.6		
1E12*MOVF053A		1KJB*Z3A		18.7		
F053B		1KJB*Z3B		18.7		
F023	(b)	1KJB*Z19, 1RBB*Z15A		36.3		
F008		1KJB*Z20		29.7		
F009	(b)	"		25.3		
F037A		1KJB*Z21A		73.7		
F037B		" B		74.8		
1E33*MOVF008	(d)	1KJB*Z1A,B,C,D		N/A		

3/4 6-20 29

8-15-84

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
1. Primary Containment (Continued)						
1G33*MOV Fo28	1KJB*Z4	7	1,2,3 AND ##	20.9	Yes	6.31
" Fo40	" Z6	7		24.2	No	
" Fo41(b)(c)	" Z7	7		19.8		
" Fo53	Z129	7		6.5		
1G33*MOV Fo34(h)	1KJB*Z4	7		20.9	Yes	
Fo39	" Z6	7		24.2	No	
Fo64 (i)	" Z7	7		6.6		
Fo54	" Z129	7		5.5		
1NCS*MOV 178	1KJB*Z5	11		12.1	Yes	
" 172	" Z5	11		12.6		
1E22*MOV Fo23	" Z11	1		50	No	
1E12*MOV Fo24A	" Z24A	10		63.8		
" Fo11A	" Z24A	10		34.1		
1E21*MOV Fo12	" Z24A	10		57.2		
1E12*MOV Fo24B	" Z24B	10		63.8		
" Fo11B	" Z24B	10		30.8		
" Fo21	" Z24C	10		97.9		
1SFC*MOV 119	" Z24	1		68		
" 120	" Z27	1		62.7		
" 122	" Z27	1		63.8		
" 139	" Z28	1		39.6		
" 121	" Z28	1		39.6		

8-15-84

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
1. Primary Containment (Continued)						
1DFR*AOV102(b)	1KJB*Z35		1,2,3 AND ##	NA	No	6.31
1DFR*AOV101(b)	"Z35					
1DER*AOV127(b)	"Z38					
1DER*AOV126(b)	"Z38					
1FPW*MOV121	"Z41			34.1	Yes	
1SAS*MOV102	"Z44			22.0		
1TAS*MOV106	"Z46			18.7		
1CCP*MOV138	"Z48			22.0	No	
"158	"Z49			23.1		
"159	"Z49			24.2		
1SWP*MOV5A	"Z53A			50.6		
"58	"Z53B			53.9		
1HVN*MOV102	"Z131			31.9	Yes	
"128	"Z131			28.6		
"127	"Z132			27.5		
1CNS*MOV125	"Z134			22.0		

8-15-84

~~DE-STS (ONR/07)~~
River Gend-Unit 1

3/4 6-30 32

8-15-84

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No) (f)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
1. Primary Containment (Continued)						
1E51-MONF063 (b)	1	1KJB4 Z15	1,2,3 AND ##	29	Z	6.31
F076 (b)	2	Z15		13.4		
F064	2	Z15		9.9		
F031	2	Z16		21.8		
F077	2	Z17		14.2		
F078	3	Z18 B,C		16.5		
1NR-AON165	8	Z31		3		
123	8	Z31		3		
128	8	Z33		3		
166	8	Z33		3		
1SSR4-SC1130	10	1KJB4 Z6018		3		
1121	10	"		3		

TABLE 3.6.[illegible]

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
2. Drywell						
1HVR * AOV 147	1DRB * Z32		1, 2, and 3	3	No	N/A
1CCP * MOV 142	" Z50			30		
" MOV 144	" Z51			30		
" MOV 143	" Z51			30		
1SWP * MOV 4A	" Z54			52.8		
" 4B	" Z54			51.7		
" 5A	" Z55			50.6		
" 5B	" Z55			53.9		
1RCS * MOV 58A	" Z152			11.0		
" 59A	" Z153			10.6		
" 60A	" Z154			6.3		
" 61A	" Z155			8.6		
" 58B	" Z156			10.6		
" 59B	" Z157			10.8		
" 60B	" Z158			6.38		
" 61B	" Z159			8.9		
1HVR * AOV 125	" Z32			3		
" 126	" Z34			3		
" 148	" Z34			3		

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No) (f)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
2. Drywell (Continued)						
1CPM*MOV 2A	1DRB*Z57A	10	1,2, and 3	30	No	N/A
" 4A	" Z57A	10		30		
" 2B	" Z57B	10		30		
" 4B	" Z57B	10		30		
" 3A	" Z58A	10		30		
" 1A	" Z58A	10		30		
" 3B	" Z58B	10		30		
" 1B	" Z58B	10		30		
1B33*AOV F019	1DRB*Z449	9		(6)	NO	
1B33*AOV F020	1DRB*"	9		(6)	NO	

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
b. Manual Isolation Valves				
1. Primary Containment				
1E12 * F099A	1KJB * Z21A	No	6.31	1,2,3 AND ##
" F099B	" Z21B			
1HVR * V8	" Z602A			
" V10	" Z602B			
1LSV * V64	" Z602D			
1HVR * V1R	" Z602F			
1LMS * V14	" Z603A			
1LMS * V12	" Z603A			
1LMS * V7	" Z603C			
1LMS * V16	" Z603C			
1CMS * V2	" Z605A			
" V3	" Z605B			
1HVR * V14	" Z606A			
" V16	" Z606B			
1CMS * V16	" Z606C			
" V15	" Z606D			
1LSV * V65	" Z606E			
1HVR * V18	" Z606F			
1E12 * VF044A	" Z21A			
1E12 * VF044B	" Z21B			

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
b. Manual Isolation Valves				
1. Primary Containment (Continued)				
1F WS * MOV 7A (e)	1M JB * Z 3A	YES	6.31	1, 2, 3 AND ##
7B (e)	Z 3B			
1E 22 * MOV F015 (e)	Z 8	NO		
F004 (b)(e)	Z 9, 10RB * Z10			
F012 (e)	Z 11			
1E 21 * MOV F001 (e)	Z 12			
F005 (b)(e)	Z 13, 10RB * Z19			
1E 51 * MOV F068 (e)	Z 17			
F019 (e)	Z 18A			
F013 (b)(e)	Z 19, 10RB * Z30			
1E 12 * MOV F021A (e)	Z 21A			
F042A (e)	Z 21A			
F021B (e)	Z 21B			
F042B (e)	Z 21B			
F042C (e)	Z 21C			
F013A (e)	Z 23A			
F013B (e)	Z 23B			
F064A (e)	Z 24A			
1E 21 * MOV F011 (e)	Z 24A			
1E 12 * MOV F064B (e)	Z 24B			
F064C (e)	Z 24C			
F004A (e)	Z 25A			
F004B (e)	Z 25B			
F105 (e)	Z 25C			

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
b. Manual Isolation Valves				
1. Primary Containment (Continued)				
1CIS*MOV083 (e)	1	NO	6.31	1,2,3 and ##
1CPP*MOV104 (e)	229			
105 (e)	233			
1SWP*MOV507A (e)	233			
507B (e)	252A			
81A (e)	252B			
81B (e)	253A			
503A (e)	253B			
503B (e)	253A			
1SW*MOV1B (e)	253B			
1A (e)	202			
1CPP*SOV140 (k)	203			
1CMS*SOV35D (e)	231			
31B (e)	260 1E			
35B (e)	260 1E			
31D (e)	260 1F			
35C (e)	260 1F			
31A (e)	260 5E			
35A (e)	260 5E			
31C (e)	260 5F			
	260 5F			

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No) (f)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
b. Manual Isolation Valves				
2. Drywell				
1SAS*V489	1DRB*Z45	NO	N/A	1, 2, AND 3
1IAS*V79	" Z47			
1HVN*V542	" Z54			
1SWP*V205	" Z54			
" V206	" Z55			
1SVV*V50	" Z107			
" V53	" Z112			
1IAS*V237	" Z166			
" V238	" Z166			
1RCS*N132	" Z152			
" N131	" Z153			
" N162	" Z154			
" V156	" Z155			
" N187	" Z156			
" N186	" Z157			
" N217	" Z158			
" N211	" Z159			
1B21*MOVF005 (e)	" Z161			
1CMS*SOV34F (e)	" Z164			
" B34E (e)	" Z165			

~~GE-STS-(BWR/6)~~
River Bend-Unit 1

3/4 6-22 39

8-15-84

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(1) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c. Other Isolation Valves				
1. Primary Containment				
1B21*AOV F032A(c)	1KJB*Z23A	Yes	6.94	1,2,3 AND ##
" VF010A(b)	" Z23A,			
" F032B(c)	" Z23B			
" VF010B(b)	" Z23B			
1E22*AOV F005(b)(c)	" Z9, 1DRB*Z10	No	6.31	
" RV F014	" Z11		6.31 (a)	
" RV F035	" Z11			
" RV F039	" Z11			
1E21*AOV F006(b)(c)	" Z13, 1DRB*Z14		6.31	
1E51*AOV F065(b)(c)	" Z19, 1DRB*Z180			
" F066(b)(c)	" Z19, "			
1E12*AOV F041C(b)(c)	" Z21C, 1DRB*Z22C			
1RHS*RV3A	" Z23A		6.31 (a)	
1E12*RV F055A	" Z23A			
" F025A	" Z23A			
" F017A	" Z23A			
" F005	" Z23A			
1E21*RV F018	" Z23A			
" F031	" Z23A			

GE-STS (BWR/6)
River Bend-Unit 1

3/4 6-32 40

8-15-84

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No) (f)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c. Other Isolation Valves				
1. Primary Containment				
1RHS*RV38	1KJB*Z23B	No	6.31	1,2,3 and ##
1E12*RVF055B	"Z23B			
F025C	"Z23B			
F025B	"Z23B			
F030	"Z23B			
F101	"Z23B			
FD17B	"Z23B			
1SFC*V101	"Z26			
1C11*VF122	"Z29			
1FPW*V203	"Z41	Yes		
1SAS*V486	"Z44			
1IAS*V80	"Z46			
1CCP*V118	"Z48	No		
1SWP*V174	"Z52A			
1SWP*V175	"Z52B			
1SVV*V9	"Z102			
1SVV*V31	"Z103			
1HVN*V541	"Z132	Yes		
1CNS*V84	"Z134			
1DFR*V180	"Z35	No		

~~GE-STS (BWR/6)~~
River Bend - Unit 1

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No) (f)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c. Other Isolation Valves				
2. Drywell				
1B21*RVF047A	1DRB4 Z136	No	N/A	1, 2, AND 3
41A	Z137			
51G	Z138			
41L	Z139			
47C	Z140			
41G	Z141			
51C	Z142			
41C	Z143			
47B	Z144			
41B	Z145			
51B	Z146			
41F	Z147			
47F	Z148			
41D	Z149			
47D	Z150			
51D	Z151			

3/4 6-37a 41

8-15-84

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (1) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c. Other Isolation Valves				
2. Drywell				
1E12 * AOV F041A (c)	1DRB * Z22A	No	N/A	1, 2 AND 3
" " F041B (c)	" Z22B			
1DFR * V4	1DRB * Z37A			
" V3	" Z37A			
" V1	" Z37B			
" V2	" Z37B			
1DER * V14	" Z40A			
" V15	" Z40A			
" V16	" Z40B			
" V17	" Z40B			
1SAS * V487	" Z45			
1IAS * V78	" Z47			
1CCP * V119	" Z50			
1SWP * RV119	" Z54			
1CAI * VEX F004A	" Z56			
" F004B	" Z56			
" VF006	" Z56			
" VF007	" Z56			
1CCP * N133	" Z51			

3/4 6-37

8-15-84

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c. Other Isolation Valves				
2. Drywell				
1B21*F036A	1DRB*Z107	No	N/A	1, 2, and 3
36F				
36G				
36P				
VF039C				
39H				
39K				
39S				
1B21*F036J	Z112			
36L				
36M				
36N				
36R				
VF039B				
39D				
39E				
1B33*F013A	Z133			
F017A	Z133			
F013B	Z135			
F017B	Z135			

3/4 6-37 43

8-15-84

Table 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

Notes

(a) Not subject to Type C leakage tests. Valve(s) will be included in the Type A test.

(b) Also isolates the drywell

(c) TESTABLE CHECK VALVE.

(d) OPENS ON ISOLATION SIGNAL.

(e) RECEIVES A REMOTE MANUAL ISOLATION SIGNAL.

(f) THIS LINE IS SEALED BY THE PENETRATION VALVE LEAKAGE CONTROL SYSTEM (PVLCS)

(g) This valve sealed by the main steam positive leakage control system (MS-PLCS)

(h) ALSO ISOLATES ON HIGH NONREGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE (RWCU)

(j) VALVES G33*MOVF001 & F004 ARE THE ONLY VALVES FROM GROUP 7 THAT ISOLATE ON THE STANDBY LIQUID CONTROL SYSTEM INITIATION SIGNAL

** When handling irradiated fuel in the ~~secondary~~ PRIMARY containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

Description of Change:

RB-U1 design does not include vacuum breakers for the free standing steel containment.

Justification:

SER (DRAFT) 6.2.1.5, page 6-10 and 6.2.1.5.2, page 6-11. Containment ventilation system unit coolers are automatically stopped on low vacuum in containment IAW Tech Spec 3/4.3.9. Question 480.13 refers.

CONTAINMENT SYSTEMS

~~3/4.6.5 SECONDARY CONTAINMENT~~

SECONDARY CONTAINMENT ~~INTEGRITY~~ – FUEL BUILDING

LIMITING CONDITION FOR OPERATION

3.6.~~5~~.1.2 ~~SECONDARY CONTAINMENT INTEGRITY~~ SECONDARY CONTAINMENT – FUEL BUILDING shall be maintained.

APPLICABILITY: ~~OPERATIONAL CONDITIONS 1, 2, 3 and *.~~ OPERATIONAL CONDITION *.

ACTION:

Without ~~SECONDARY CONTAINMENT INTEGRITY~~ SECONDARY CONTAINMENT – FUEL BUILDING,

~~a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

~~b.~~ In ~~Operational Condition *~~ FUEL BUILDING, suspend handling of irradiated fuel in the ~~secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel.~~ The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

~~4.6.5.1.2 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:~~

INSERT A – ~~a. Verifying at least once per 24 hours that~~ The pressure within the ~~secondary containment~~ FUEL BUILDING is less than or equal to ~~0.25~~ inches of vacuum water gauge.

~~b. Verifying at least once per 31 days that:~~

b. ~~1.~~ All ~~secondary containment~~ FUEL BUILDING equipment hatches ~~and blowout panels~~ are ~~closed and sealed~~.

c. ~~1.~~ ~~(At least one)~~ INSTALLED. The door in each access to the secondary containment is closed, except for routine entry and exit.

d. ~~1.~~ All ~~secondary containment~~ FUEL BUILDING penetrations ~~not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and~~ required to be closed during accident conditions are closed by valves, blind flanges, or ~~deactivated automatic~~ dampers ~~valves~~ secured in position. FUEL HANDLING

~~c. At least once per 18 months:~~

~~1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to (0.25) inches of vacuum water gauge in less than or equal to (120) seconds, and~~

~~2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to (0.25) inches of vacuum water gauge in the secondary containment at a flow rate not exceeding (2300) CFM.~~

*When irradiated fuel is being handled in the ~~secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.~~ FUEL BUILDING.

EXCEPT THE FUEL BUILDING VENTILATION SYSTEM CHARCOAL FILTRATION SUBSYSTEM FLOW PATH PENETRATIONS;

INSERT A

4.6.5.1.2 SECONDARY CONTAINMENT — FUEL BUILDING SHALL BE DEMONSTRATED WITHIN 24 HOURS PRIOR TO AND AT LEAST ONCE PER 7 DAYS DURING HANDLING OF IRRADIATED FUEL IN THE FUEL BUILDING BY VERIFYING THAT:

INSERT EOC
3/4 6-46

10-8-84

Title, Applicability and Action - All sections made specific and limited to operational conditions 1, 2 & 3. Deleted requirements applicable to handling of irradiated fuel. RB-U1 design presently allows handling of irradiated fuel only in fuel bldg. and in primary containment. Specification 3/4.6.5.1.2 requires fuel building secondary automatic isolation dampers, except for FBVS charcoal filtration subsystem flow path, to be closed but not operable. Made system plural because there are several ventilation systems title, LCO 3.6.5.2, action and surveillance requirement 4.6.5.2
Nc valves were identified.

Action b&c - Added footnote*. Similar footnote in GE-STS 3/4.6.4. Should be permitted for dampers, also.

SR 4.6.5.2.c - Dampers will not be in ASME pump and valve ISI program. Moved isolation time test to SR 4.6.5.2.b.

Table 3.6.5.2-1 - Added damper group column and footnote* for clarity with Table 3.3.2-1.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION ~~DAMPERS/(VALVES)~~

~~DAMPER/(VALVE)~~ FUNCTION	MAXIMUM ISOLATION TIME (Seconds)	DAMPER GROUP*
1. A SHIELD Annulus Reactor Building Ventilation EXHAUST ~~Supply (Damper)(Valve)~~ (1HVR*AOD1A)	~~(5)~~ 15	12
2. A SHIELD Annulus Reactor Building Ventilation EXHAUST ~~Supply~~ ~~(Damper)(Valve)~~ (1HVR*AOD2A)	~~(5)~~ 15	12
3. A SHIELD Annulus Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (1HVRD AOD22B)	~~(5)~~ 15	12
4. A Auxiliary Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (1HVR*AOD21A)	~~(5)~~ 15	11
5. A Auxiliary Reactor Building Ventilation EXHAUST ~~Supply (Damper)(Valve)~~ (1HVR*AOD26A)	~~(5)~~ 15	11
6. A Auxiliary Reactor Building Ventilation ~~Supply~~ EXHAUST ~~(Damper)(Valve)~~ (1HVR*AOD24A)	~~(5)~~ 15	11
7. A Auxiliary Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (1HVR*AOD10A)	~~(5)~~ 15	11
8. A Auxiliary Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (1HVR*AOD10B)	~~(5)~~ 15	11
9. A Auxiliary Reactor Building Ventilation Supply ~~(Damper)(Valve)~~ (1HVR*AOD14A)	~~(5)~~ 15	11
10. A Auxiliary Reactor Building Ventilation Supply ~~(Damper)(Valve)~~ (1HVR*AOD16A)	~~(5)~~ 15	11
11. A Fuel Reactor Building Ventilation ~~Exhaust~~ SUPPLY (Damper)(Valve) (1HVF*AOD18A)	~~(5)~~ 15	13
12. A Fuel Reactor Building Ventilation ~~Exhaust~~ SUPPLY ~~(Damper)(Valve)~~ (1HVF*AOD1B)	~~(5)~~ 15	13
13. A Fuel Reactor Building Ventilation Supply ~~(Damper)(Valve)~~ (1HVF*AOD1A)	~~(5)~~ 15	13
14. A Fuel Reactor Building Ventilation Supply ~~(Damper)(Valve)~~ (1HVF*AOD13A)	~~(5)~~ 15	13
15. A Fuel Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (1HVF*AOD10A)	~~(5)~~ 15	13
16. A Fuel Reactor Building Ventilation Exhaust ~~(Damper)(Valve)~~ (1HVF*AOD11A)	~~(5)~~ 15	13

~~5~~ SEE TABLE 3.6.2-1.

(~~The~~ provisions of Specification 3.0.4 are not applicable.)

8-15-84

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing testing acceptance criteria of less than ~~(1%)~~ 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 system flow rate is ~~(2300)~~ (12,500) cfm ± 10%.

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, for a methyl iodide penetration of less than ~~(1%)~~ 0.175%; and

3. Verifying a subsystem flow rate of ~~(2300)~~ (12,500) cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, for a methyl iodide penetration of less than ~~(1%)~~ 0.175%.

d. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, for the:

a) LOCA, and

b) ~~Fuel handling accident.~~ High radiation signal

2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~(8)~~ inches Water Gauge while operating the filter train at a flow rate of ~~(2300)~~ (12,500) cfm ± 10%.

3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

a. Manual initiation from the main control room, and

b. Simulated automatic ~~initiation~~ initiation signal.

~~(4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.)~~

~~5. Verifying that the heaters dissipate (9.2) ± (1.0) kw when tested~~ ≥ 61 in accordance with ANSI N510-1975.

NOTE: THE NRC STAFF HAS BEEN REQUESTED TO REVISE THE SER IN ACCORDANCE WITH INFORMATION CONTAINED IN THE FSAR ON THE CHARCOAL ABSORBER EFFICIENCY OF 99%.

INSERT FOR PAGE 3/4 6-52

a. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

1. LOCA, and
2. High Radiation signal

b. Verifying a subsystem flow rate of (52,500) cfm ± 10% during system operation.

c. Verifying that the subsystem starts, the SGTS inlet valve opens and the automatic pressure control outlet valve closes on each of the following test signals:

1. Manual initiation from the control room, and
2. Simulated automatic initiation signal.

9-6-84

SR 4.6.5.5.1

a. RB-U1 Terminology

b, b.2, c, e, f - Punctuation changes and phrase addition per NRC "Revised Surveillance Requirements for testing HEPA Filters and Charcoal Adsorber Units".

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing ~~testing~~ acceptance criteria of less than ~~(1)~~ 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 system flow rate is ~~(2300)~~ (10,000) cfm ± 10%.

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, for a methyl iodide penetration of less than ~~(1)~~ 0.175%; and

3. Verifying a subsystem ~~flow rate of (2300)~~ (10,000) cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, for a methyl iodide penetration of less than ~~(1)~~ 0.175%.

d. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

   a) LOCA, and

   b) Fuel handling accident.

2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~(8)~~ inches Water Gauge while operating the filter train at a flow rate of ~~(2300)~~ (10,000) cfm ± 10%.

3. Verifying that the ~~filter train~~ SUBSYSTEM starts, and isolation dampers ~~open~~ on each of the following test signals:

   a. Manual initiation from the control room, and ACTUATE TO ISOLATE THE NORMAL FLOW PATH AND TO DIVERT FLOW THROUGH THE CHARCOAL FILTRATION SUBSYSTEM

   b. Simulated automatic ~~initiation~~ INITIATION signal.

~~4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.~~

4. Verifying that the heaters dissipate ~~(9.3)~~ 5.7 ± ~~(1.0)~~ 5.7 kw when tested in accordance with ANSI N510-1975.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT/~~AND~~ DRYWELL HYDROGEN MIXING SYSTEM - DRYWELL PRESSURE CONTROL

LIMITING CONDITION FOR OPERATION

3.6.6.2 ~~Two independent~~ ONE PRIMARY containment/~~and~~ drywell hydrogen mixing system~~s~~/~~shall be OPERABLE~~. INLET LINE MAY BE OPEN FOR DRYWELL PRESSURE CONTROL PROVIDED THAT PRIMARY CONTAINMENT PURGE VALVES ARE CLOSED.

APPLICABILITY: OPERATIONAL CONDITIONS 1, ~~and~~ 2 AND 3.

ACTION:

INLET LINE AND PRIMARY CONTAINMENT PURGE VALVE POSITION NOT SATISFIED,

With ~~one~~ containment/~~and/or~~ drywell hydrogen mixing system ~~inoperable~~, restore the ~~inoperable system to OPERABLE status within 30 days~~ or be in at least HOT SHUTDOWN within the next 12 hours AND IN COLD SHUTDOWN WITHIN THE FOLLOWING 24 HOURS.

THE ABOVE RESTRICTIONS ON PRIMARY INLET LINE AND PURGE VALVES TO WITHIN THE RESTRICTIONS OR OTHERWISE ISOLATE THE PENETRATIONS WITHIN 4 HOURS

SURVEILLANCE REQUIREMENTS

4.6.6.2 WITHIN ONE HOUR PRIOR TO OPENING A PRIMARY CONTAINMENT/ ~~Each containment and drywell hydrogen mixing system shall be demonstrated OPERABLE~~:

DRYWELL HYDROGEN MIXING SYSTEM INLET LINE, VERIFY THAT THE ABOVE REQUIRED VALVE POSITION RESTRICTIONS ARE SATISFIED)

~~a. At least once per 92 days by~~

~~1. Starting the system from the control room, and~~

~~2. Verifying that the system operates for at least 15 minutes.~~

~~b. At least once per 18 months by verifying a system flow rate of at least _____ cfm.~~

4.6.6.2.2 Once per 24 hours verify the above restrictions are satisfied.

Description of Change:

RB-U1 design provides for a drywell purge system, but considers it a non-ESF, backup system. FSAR 6.2.5.1(5), Page 6.2-76; 6.2.5.1(10), Page 6.2-77; 6.2.5.2(3), Page 6.2-78; 6.2.5.2.3, Page 6.2-82; 6.2.5.3.4, Page 6.2-86; Table 6.2-33; SER (Draft) 6.2.5, Page 6-44 refers.

DELETED

DRYWELL PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.3 Two independent drywell purge system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell purge subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.3 Each drywell purge system subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Starting the subsystem from the control room, and
2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by:

1. Verifying a subsystem flow rate of at least (500) cfm during subsystem operation for at least 15 minutes.
2. Verifying the pressure differential required to open the vacuum breakers on the drywell purge compressor discharge lines, from the closed position, to be less than or equal to (1.0) psid.

c. Verifying the OPERABILITY of the drywell purge compressor discharge line vacuum breaker isolation valve differential pressure actuation instrumentation with the opening setpoint of (1.0) psid by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

NOTE: THIS SPEC IS UNDER REVISION

GE-STS

for hydrogen recombiners allows 30 days out-of-service time, so it is consistent to allow one igniter to be cut-of-service for 30 days also.

Added exception to Specification 3.0.4 hydrogen recombiners redundant to hydrogen igniters.

SR 4.6.7.3

a - Current limit should be determined during pre-op testing. Number of igniters required operable per train should be determined.

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the ~~following~~ independent standby service water (SSW) system subsystems ASSOCIATED WITH SYSTEMS AND COMPONENTS REQUIRED TO BE OPERABLE, with each subsystem comprised of:

a. ~~One~~ Two OPERABLE SSW pumps, and

b. An OPERABLE flow path capable of taking suction from the STANDBY COOLING TOWER BASIN ~~turbine-building heat sink~~ and transferring the water through the ~~RHR heat exchangers, ECCS pump room cooler coolers, and associated coolers and pump heat exchangers~~ ASSOCIATED SYSTEMS AND COMPONENTS REQUIRED TO BE OPERABLE,

shall be OPERABLE.

~~a. In OPERATIONAL CONDITION 1, 2 and 3, two subsystems.~~

~~b. In OPERATIONAL CONDITION 4, 5 and *, the subsystem(s) associated with systems and components required OPERABLE by Specifications 3.4.9.1, 3.4.9.2, (3.5.2), 3.9.11.1, 3.9.11.2 and 3.8.1.2.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

INSERT A → b.

INSERT B → a. In OPERATIONAL CONDITION 1, 2 or 3:

With one SSW subsystem OTHERWISE inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~b.~~ With both SSW subsystems OTHERWISE inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN** within the following 24 hours.

~~b. In OPERATIONAL CONDITION 3 or 4 with the SSW subsystem inoperable which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, as applicable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.~~

~~c. In OPERATIONAL CONDITION 4 or 5 with the SSW subsystem inoperable which is associated with an ECCS pump required OPERABLE by Specification 3.5.2, declare the associated ECCS pump inoperable and take the ACTION required by Specification 3.5.2.~~

*When handling irradiated fuel in the ~~Auxiliary Building or Containment Building~~ PRIMARY OR SECONDARY CONTAINMENT.

**Whenever both RHRSW ~~subsystems~~ are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

INSERT B

1. With one SSW pump and/or cell from the same subsystem inoperable, restore the inoperable pump and/or cell to operable status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With two SSW pumps and/or cells from one SSW subsystem inoperable, restore the inoperable pumps and/or cells to operable status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one SSW pump and/or cell from each SSW subsystem inoperable, restore the inoperable pumps and/or cells to operable status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

INSERT A

a. With one or more of the required OPERABLE flow path(s), pump(s) or associated systems and/or components required to be OPERABLE, declare the associated system(s) and/or component(s) inoperable and take the required action.

SR 4.7.1.1

b.1 - FSAR 7.3.1.1.8(2), page 7.3-25; 9.2.7.5, page 9.2-45; 8.3.1.1.6.2(3), page 8.3-42.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

~~d. In OPERATIONAL CONDITION 5 with the SSW subsystem inoperable which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2.~~

~~e. In Operational Condition 3, with the SSW subsystem inoperable which is associated with a diesel generator required OPERABLE by Specification 3.8.1.2, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2.~~

INSERT B →

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system subsystem(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by verifying that ~~each~~ automatic valve servicing safety related equipment actuates to the correct position on ~~a test signal~~ A REACTOR PLANT COOLING WATER SYSTEM (RPCCW) LOW PRESSURE SIGNAL AND ON A NORMAL SERVICE WATER LOW PRESSURE SIGNAL.

OR ISOLATING NON-SAFETY RELATED EQUIPMENT

1 EACH

2. ONE PUMP IN EACH SUBSYSTEM STARTS ON A:
   a) RPCCW LOW PRESSURE SIGNAL, AND
   b) NORMAL SERVICE WATER LOW PRESSURE SIGNAL

3. EACH PUMP IN EACH SUBSYSTEM STARTS ON A MANUAL CONTROL SIGNAL FROM THE MAIN CONTROL ROOM

INSERT B

C. IN OPERATIONAL CONDITIONS 4, 5 AND *, WITH THE SSW SUBSYSTEM(S), WHICH IS ASSOCIATED WITH SYSTEMS AND/OR COMPONENTS REQUIRED TO BE OPERABLE, OTHERWISE INOPERABLE, DECLARE THE ASSOCIATED SYSTEMS AND COMPONENTS INOPERABLE AND TAKE THE REQUIRED ACTION. . THE PROVISIONS OF SPECIFICATION 3.0.3 ARE NOT APPLICABLE IN OPERATIONAL CONDITION *.

8-15-84

Description of Change:

Deleted Tech Spec.

Justification:

Standby Service Water System Services HPCS System in RB-U1 design.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation ~~and restart~~ and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.

2. Verifying that the system will develop a flow of greater than or equal to ~~600~~ gpm in the test flow path when steam is supplied ~~to the turbine~~ at a reactor pressure of ~~150~~ + ~~15~~ - ~~0~~ psig.*

~~3~~. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal and on a suppression pool water level - high signal.~~

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

NOTE: THIS TECHNICAL SPECIFICATION WILL BE PROVIDED AT A LATER DATE DUE TO THE EVALUATION OF RECENT NRC LETTERS.

PLANT SYSTEMS

3/4.7.7 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The fire suppression water system shall be OPERABLE with:

a. THREE fire suppression pumps, each with a capacity of 1500 gpm, with their discharge aligned to the fire suppression header,

b. TWO separate fire water TANKS ~~supplies~~, each with a minimum NET AVAILABLE ~~contained~~ volume of 265,000 gallons, and

c. An OPERABLE flow path capable of taking suction from BOTH WATER STORAGE ~~the~~ tank ~~and the tank~~ 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 ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.7.2, 3.7.7.4, and 3.7.7.5.

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 suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the minimum contained water supply volume.

b. At least once per 31 days ~~(on a STAGGERED TEST BASIS)~~ by starting ~~(each)~~ the electric motor driven fire suppression pump and operating it for at least 15 minutes on recirculation flow.

c. 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.

~~(d. At least once per 6 months by performance of a system flush.)~~

f.2 - FSAR 9.5.1.2, page 9.5-2; 9.5.1.2.2, page 9.5-3.

f.4 - FSAR 9.5.1.2.2, page 9.5-3.

SR 4.7.7.1.2 FSAR 9.5.1.2, page 9.5-2.

SURVEILLANCE REQUIREMENTS (Continued)

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

f. 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 automatic valve in the flow path actuates to its correct position,

2. Verifying that each fire suppression pump develops at least ( ) gpm at a system head of ( ) feet,

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

4. Verifying that each fire suppression pump starts ~~(~~sequentially~~)~~ to maintain the fire suppression water system pressure greater than or equal to 70 psig.

g. 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.

4.7.7.1.2 ~~(The)~~ (Each) diesel driven fire suppression pump shall be demonstrated OPERABLE:

a. At least once per 31 days by;

1. Verifying the fuel day tank contains at least 300 gallons of fuel.

2. Starting the pump from ambient conditions and operating for greater than or equal to 30 minutes on recirculation 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-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

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.

SR 4.7.7.1.3 FSAR 9.5.1.2, page 9.5-2.

a.1,2 - Pilot cells will be designated.

a.3 - Deleted. No way to measure cell voltage.

c.1 - Many cell plates not visible.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.4 The ~~following~~ Halon systems protecting the main control room Power Generation Control Complex (PGCC) shall be OPERABLE with the storage tanks having at least 95% of full charge ~~(level)~~ weight and 90% of full charge pressure.

~~a. (Plant dependent - to be listed by name and location.)~~

~~b.~~

~~c.~~

APPLICABILITY: Whenever equipment protected by the Halon systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish ~~a continuous fire suppression equipment for hose or components could be damaged fire~~ with backup fire ~~equivalent system or establish~~ an hourly fire watch patrol for the PGCC.

b. The provisions of Specification 3 [illegible] not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.3 ~~Each of~~ The above required Halon 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 6 months by verifying Halon storage tank weight and pressure ~~(level)~~.

c. At least once per 18 months by:

1. Verifying the system, ~~including associated ventilation system fire dampers and fire door release mechanisms~~, actuates, manually and automatically, upon receipt of a simulated actuation signal, ~~and actual release of Halon~~ may be excluded from this test, AND

2. ~~Performance of a flow test through (accessible) headers and~~ nozzles to assure no blockage. AN AIR

~~(Accessible headers and nozzles.)~~

PLANT SYSTEMS

4.7.11 STRUCTURAL SETTLEMENT

LIMITING CONDITION FOR OPERATION

3.7.11 Structural settlement of the following structures shall be within the predicted values.

a. Reactor Building
b. Auxiliary Building
c. Fuel Building
d. Control Building
e. Diesel Generator Building
f. Standby Cooling Tower, Basin and Pump House

APPLICABILITY: At all times.

ACTION:

With the measured structual settlement of any of the above required structures outside of the predicted settlement, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the settlement measurements and the predicted settlement, an analysis to demonstrate the continued structural integrity of the affected structure(s) and plans to monitor the settlement of the affected structure(s) in the future.

SURVEILLANCE REQUIREMENTS

4.7.11 The structural settlement of the above required structures shall be demonstrated to be within the predicted settlement values:

a. At least once per 92 days, using at least three markers per structure, until there is essentially no movement during those 92 days.

b. At least once per 24 months, using at least one marker per structure, for at least 10 years.

c. Following any seismic event equal to or greater than an Operational Basis Earthquake (OBE), using at least three markers per structure.

LCO 3.8.1.1

b.1 - FSAR 8.3.1.1.3.6.1.1, p. 8.3-11; 8.3.1.1.3.6.2.1, p. 8.3-16; 8.3.1.2.1.2, p. 8.3-44; 8.3.1.2.2.1(1.6), p. 8.3-48; 9.5.4, p. 9.5-17; 9.5.4.1 (6)(10), p. 9.5-18; 9.5.4.2(4), p. 9.5-19; 9.5.4.3, p. 9.5-20a.

b.2 - FSAR 8.3.1.1.3.6.1.1, p. 8.3-11; 8.3.1.1.3.6.1.2, p. 8.3-16; 8.3.1.2.1.2, p. 8.3-45; 9.5.4.2(1)(2), p. 9.5.18, 18a, 19; 9.5.4.3, p. 9.5.20.
Equivalent level indicated on instrumentation added.

Action

a, b, c, d, f - 1A, 1B and 1C - Correct RB-U1 Terminology.

a, b, c, e - Time to demonstrate operability of remaining A.C. sources relaxed by guidance in NCR Item 5. GSU interprets that guidance as permitting one additional hour to test each Diesel Generator that must be demonstrated operable.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

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. Three separate and independent diesel generators, each with:

1. A ~~3~~ separate day ~~and engine mounted fuel~~ tanks containing a minimum of ~~(250)~~ 316.3 gallons of fuel, EQUIVALENT TO A LEVEL OF (—)%
2. A separate fuel storage ~~system~~ TANK containing a minimum of ~~(26,000)~~ 45,499 gallons of fuel, and EQUIVALENT TO A LEVEL OF (—)%,
3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator (1A) or (1B) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one ~~hour~~ WITHIN ONE HOUR and at least once per 8 hours thereafter; restore at least two offsite circuits and diesel generators (1A) and (1B) to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. ADDITIONAL HOUR FOR EACH DIESEL GENERATOR TESTED,

b. With one offsite circuit and diesel generator (1A) or (1B) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one ~~hour~~ WITHIN ONE HOUR and at least once per 8 hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators (1A) and (1B) to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ADDITIONAL HOUR FOR EACH DIESEL GENERATOR TESTED)

ACTION (C) - DELETED. DIVISIONAL DIESELS (A+B) SHOULD NOT BE REQUIRED IF HPCS DIESEL IS INOPERABLE. THE INTENT OF THIS ACTION (TO FIND GENERIC PROBLEMS) IS NOT APPLICABLE BECAUSE THE HPCS AND STANDBY DIESELS ARE OF DIFFERENT MANUFACTURER

NOTE - (d): To be deleted with addition of SPECIFICATION 3.0.5 (TO BE DISCUSSED WITH NRC STAFF)

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With diesel generator /1C/ of the above required A.C. electrical power sources inoperable, ~~demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter;~~ restore the inoperable diesel generator /1C/ to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

d. With diesel generator /1A/, /1B/ or /1C/ of the above required A.C electrical power sources inoperable, in addition to ACTION a, b or c, as applicable, verify within 2 hours 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; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performin Surveillance Requirement 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter, FOR EACH DIESEL GENERATOR TESTED, unless the diesel generators are already operating; restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

f. With diesel generators /1A/ and /1B/ of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators /1A/ and /1B/ to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators /1A/ and /1B/ to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SR 4.8.1.1.2

a.1 - See LCO 3.8.1.1.b.1

a.3 - See LCO 3.8.1.1.b.1

a.4 - Footnote * - NRC Item # 5.

10 seconds - FSAR 8.3.1.2.1.2(6), p. 8.3-45; 8.3.1.2.2.1(17), p.8.3-47; 6.2.1.1.3.1.4.2, p. 6.2-14; Tables 6.3.1, 8.3-2.

a.5 - Footnote * - NRC Item # 5.

DG ratings - 8.3.1.1.3.3, p. 8.3-8; 8.3.1.1.3.6.1.1, p. 8.3-12; 8.3.1.1.3.6.2.1, p. 8.3-15; 8.3.1.2.2.1(1.9), p. 8.3-49; Table 8.3-3.

a.7 -

FSAR 9.5.6.1, p. 9.5-27; 9.5.6.2.1, p. 9.5-29; 9.5.6.3, p. 9.5-30a.

b. - See LCO 3.8.1.1.b.1. Made tanks singular - one per diesel.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required 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 alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by ~~transferring~~, (manually) ~~and automatically~~, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day ~~and engine-mounted fuel~~ tanks.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day ~~and engine-mounted fuel~~ tanks.

4. Verifying the diesel starts from ambient condition and accelerates to at least (900) rpm in less than or equal to 10 ~~(13)~~ seconds*. The generator voltage and frequency shall be ~~(4160) ± (420) volts and (60) ± (1.2) Hz within (13) seconds after the start signal~~. The diesel generator shall be started for this test by using one of the following signals:

450 RPM FOR DIESEL GENERATORS 1A AND 1B AND FOR DIESEL GENERATOR 1C

INSERT A

a) Manual.
b) Simulated loss of offsite power by itself.
c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
d) An ESF actuation test signal by itself.

5. Verifying the diesel generator is synchronized, loaded to greater than or equal to ~~(continuous rating)~~ 3500 kw for diesel generators (1A) and (1B) and ~~(continuous rating)~~ 2600 kw for diesel generator (1C) in less than or equal to (60) seconds*, and operates with this load for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to (250) psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day ~~and engine-mounted fuel~~ tanks.

INSERT B

SURVEILLANCE REQUIREMENTS (Continued)

4. Simulating a loss of offsite power by itself, and:

a) For divisions I and II:

1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 ~~(13)~~ seconds, energizes the auto-connected ~~(shutdown)~~ loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the ~~shutdown~~ loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at (4160) ± (420) volts and (60) ± (3) ~~(1.2)~~ Hz during this test.

b) For division III:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with (the permanently connected) ~~(its)~~ loads within (13) seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the ~~shutdown~~ loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at (4160) ± (420) volts and (60) ± (1.2) Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be (4160) ± (420) volts and (60) ~~(1.2)~~ Hz within ~~(13)~~ 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6 ~~Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.~~ For diesel generator 1A and 1B, and 4160 ± 420 volts and 60 ± 1.2 Hz for diesel generator 1C

6 Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

a) For divisions I and II:

1) ~~Verifying deenergization of the emergency busses and~~ load shedding from the emergency busses.

e.6.a)2) - 10 seconds - FSAR 8.3.1.2.1.2(6), p. 8.3-45; Table 8.3.-2; Table 6.3-1.
"Shutdown" loads is not RB-U1 Terminology.

e.6.b)2) - 10 seconds - FSAR 6.2.1.1.3.1.4.2, p. 6.2-14; Table 6.3-1; 8.3.1.2.2.1(17), p. 8.3-47.

e.7 - FSAR 8.3.1.1.3.6.1.2(a.2), p. 8.3-14; 8.3.1.1.4.2, p. 8.3-33.

e.8 - 2 hour rating - FSAR 8.3.1.1.3.6, p. 8.3-10a; 8.3.1.1.3.6.1.1, p. 8.3-12; 8.3.1.1.5.3, p. 8.3-38; 8.3.1.2.2.1(1.9), p. 8.3-49,
DG-1C - 30 minute rating - 3050 kw, Table 8.3-3; 2000 hour rating - 2850 kw, FSAR 8.3.1.1.3.6.2.1, p. 8.3-15; Table 8.3-3. FSAR 8.3.1.1.5.3, p. 8.3-38
Continuous rating - FSAR 8.3.1.1.3.3, p. 8.3-8; 8.3.1.1.3.6.1.1, p. 8.3-12; 8.3.1.1.3.6.2.1, p. 8.3-15; 8.3.1.2.2.1(1.9), p. 8.3-49; 8.3.1.2.2.2(1.9, 4), p. 8.3-57; Table 8.3-3. Footnote * shows continuous ratings.

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within ~~(13)~~ 19 seconds, energizes the auto-connected ~~(shutdown)~~ loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at (4160) ± (420) volts and (60) ~~(3) (1.2)~~ Hz during this test.

b) For division III:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads and the auto-connected emergency loads within (13) seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at (4160) ± (420) volts and (60) ± (1.2) Hz during this test.

7 Verifying that all automatic diesel generator trips are automatically bypassed upon loss of voltage on the emergency bus ~~concurrent with~~ an ECCS actuation signal ~~except~~ AND, for Diesel Generator 1A and 1B, ON MANUAL EMERGENCY START, except ~~a) for divisions 1 and 2~~, engine overspeed and generator differential current).

~~b) For division 3, (engine overspeed and generator differential current).~~

8 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 greater than or equal to ~~(2-hour rating)~~ 3850 kW for diesel generators (1A) and (1B) and ~~(2-hour rating)~~ 2850 kW for diesel generator (1C). During the remaining 22 hours of this test, the diesel generator shall be loaded to ~~(continuous rating) kw~~ 3500KW for diesel generator (1A) and (1B) and ~~(continuous rating) kw~~ 2600KW for diesel generator (1C). The generator voltage and frequency shall be (4160) ± (420) volts and ~~(60) (3) (1.2)~~ Hz within 10 ~~(13)~~ seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.e.4.a)2) and b)2)*.

For Diesel Generator 1A and 1B, and 4160±420 volts and 60 ± 1.2 Hz for Diesel Generator 1C

*If Surveillance Requirements 4.8.1.1.2.e(4) a)2 and b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at ~~(continuous rating)~~ for one hour or until operating temperatures have stabilized.

3500 kw for Diesel generator 1A and 1B, and 2600 kw for Diesel Generator 1C

e.9 - DG-1A and 1B do not have a 2000-hour rating. Their continuous rating is shown FSAR 8.3.1.1.3.3, p. 8.3-8; 8.3.1.1.3.6.1.1, p. 8.3-12; 8.3.1.2.2.1(1.9), p. 8.3-49. DG-1C - FSAR 8.3.1.1.3.6.2.1, p. 8.3-15; Table 8.3-3

e.14 - Deleted. There are no installed cross connection lines. FSAR Figure 9.5-2. Subsequent sections renumbered.

e.13 - FSAR 8.3.1.1.3.6.1.2(a.1), p. 8.3-14; 8.3.1.1.4.1(1), p. 8.3-22; 8.3.1.1.4.1.2(1), p. 8.3-31; 8.3.1.2.2.2(1.47), p. 8.3-57.

Change is for clarity.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division ~~(1)~~ I, consisting of:
   1. 125 volt battery ~~(1A)~~.
   2. 125 volt full capacity charger. ← class 1E source

b. Division ~~(2)~~ II, consisting of:
   1. 125 volt battery ~~(1B)~~.
   2. 125 volt full capacity charger. ← class 1E source

c. Division ~~(3)~~ III, consisting of:
   1. 125 volt battery ~~(1C)~~.
   2. 125 volt full capacity charger. ← class 1E source

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division ~~(1)~~ I or Division ~~(2)~~ II battery and/or charger of the above required D.C. electrical power sources inoperable, restore the inoperable division to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division ~~(3)~~ III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

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

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

   2. Total battery terminal voltage is greater than or equal to ~~(129)~~ 130 volts on float charge.

c.4 - Charger nameplate and tech manual data. FSAR 8.3.2.3.1.2(1.32), p. 8.3-84; 8.3.2.4.2, p. 8.3-85, 86. Question 430.57(1.b) Q&R p. 8.3-50, 51.

d.2.a), b, c) - FSAR 8.3.2.3.1.2(1.32), p. 8.3-84; 8.3.2.4.2, p. 8.3-85, 86; Tables 8.3-4, 5, 6; Figures 8.3-7, 8, 13.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below ~~(110)~~ volts, or battery overcharge with battery terminal voltage above ~~(150)~~ volts, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than ~~(150 x 10^{-6})~~ ohms, ~~and~~ with the average of all intercell connections not exceeding 20×10^{-6} ohms, AND

3. The average electrolyte temperature of at least one out of every six ~~(a representative number) of~~ connected cells is above 60°F~~)~~.

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

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

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

3. The resistance of each cell~~(~~-to-cell~~)~~ and terminal connection is less than or equal to ~~(~~150 x 10^{-6}~~)~~ ohms, ~~and~~ with the average of all intercell connections not exceeding 20×10^{-6} ohms, AND

4. The battery charger will supply ~~at least (100) amperes~~ at a minimum of ~~(125)~~ 130 volts for at least ~~(4)~~ hours, at least: (Insert A)

d. At least once per 18 months, (Insert B) during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for the design duty cycle when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile in accordance with IEEE 450 while maintaining the battery terminal voltage greater than or equal to ~~(105)~~ volts.

(Insert C)

~~a) Battery (1A), greater than or equal to ( ) amperes; battery (1B), greater than or equal to ( ) amperes; and battery (1C), greater than or equal to ( ) amperes during the initial 60 seconds of the test.~~

~~b) Battery (1A), greater than or equal to ( ) amperes; battery (1B), greater than or equal to ( ) amperes; and battery (1C), greater than or equal to ( ) amperes during the remainder of the first hour of the test.~~

SURVEILLANCE REQUIREMENTS (Continued)

~~e) Battery (1A), greater than or equal to ( ) amperes; battery (1B), greater than or equal to ( ) amperes; and battery (1C), greater than or equal to ( ) amperes during the remainder of the (8) hour test.~~

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. At this once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. ANNUAL ~~At least once per 18 months during shutdown~~ performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating. I

and the battery current limiting characteristics are consistent with the performance data included in their procurement specification,

INSERT A

, WHILE MAINTAINING CURRENT LIMITING CHARACTERISTICS CONSISTENT WITH THE PERFORMANCE DATA INCLUDED IN THEIR PROCUREMENT SPECIFICATION,

INSERT B

a) 300 AMPERES FOR CHARGERS 1A and 1B.

b) 50 AMPERES FOR CHARGER 1C.

INSERT C

a) FOR BATTERIES 1A AND 1B:
1) DURING THE INITIAL 60 SECONDS OF THE TEST;
(a) BATTERY 1A - 671 AMPERES.
(b) BATTERY 1B - 502 AMPERES.
2) DURING THE NEXT NINE MINUTES OF THE TEST;
(a) BATTERY 1A - 270 AMPERES.
(b) BATTERY 1B - 261 AMPERES.
3) DURING THE NEXT 60 SECONDS OF THE TEST;
(a) BATTERY 1A, 336 AMPERES.
(b) BATTERY 1B, 327 AMPERES.
4) DURING THE NEXT 228 MINUTES OF THE TEST;
(a) BATTERY 1A, 270 AMPERES.
(b) BATTERY 1B, 261 AMPERES.
5) DURING THE 240TH MINUTE OF THE TEST;
(a) BATTERY 1A, 451 AMPERES.
(b) BATTERY 1B, 327 AMPERES.

b) FOR BATTERY 1C:
1) DURING THE INITIAL 60 SECONDS OF THE TEST, 77.4 AMPERES.
2) DURING THE NEXT 239 MINUTES OF THE TEST, 11.4 AMPERES.

INSERTS FOR
3/4 8-12 #13

Aug 11, 1984

f - IEEE-450(1975) requires annual test.
RB-U1 committed to IEEE-450(1975). However, IEEE-450(1975), 4,2(3) shows perform per IEEE-308(1974) which shows per fuel cycle (maximum of 18 months).

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized: ~~with tie breakers open (both) between redundant buses within the unit (and between units at the same station):~~

a. A.C. power distribution:

1. Division ~~(1)~~ I, consisting of:
   a) 4160 volt A.C. bus 1ENS*SWG1A.
   b) 480 volt A.C. ~~MCCs (and)~~ SWITCHGEAR 1EJS*SWG1A AND 1EJS*SWG2A.
   c) ~~120 volt A.C. distribution panels in (480 volt MCCs , and) (energized from inverter # connected to D.C. Division (1)* and 480 volt bus(ses)).~~

INSERT A

2. Division ~~(2)~~ II, consisting of:
   a) 4160 volt A.C. bus 1ENS*SWG1B.
   b) 480 volt A.C. ~~MCCs (and)~~ SWITCHGEAR 1EJS*SWG2B AND 1EJS*SWG2B.
   c) ~~120 volt A.C. distribution panels in (480 volt MCCs , and) (energized from inverter # connected to D.C. Division (2)* and 480 volt bus(ses)).~~

INSERT B

3. Division ~~(3)~~ III, consisting of:
   a) 4160 volt A.C. bus 1E22*S002.
   b) 480 volt A.C. MCCs SWITCHGEAR 1E22*S002.
   c) 120 volt A.C. distribution panel in 480 volt MCC 1E22-S002
   ~~(energized from inverter # connected to D.C. Division (3)* and 480 volt bus(ses)).~~ 1E22*S002 PNL

b. D.C. power distribution:

1. Division ~~(1)~~ I, consisting of 125 volt D.C. distribution panels 1ENB*PNL02A AND 1ENB*PNL03A, AND MCC 1ENB*MCC1, Bus 1ENB*SWG01A,
2. Division ~~(2)~~ II, consisting of 125 volt D.C. distribution panels 1ENB*PNL01B AND 1ENB*PNL03B Bus 1ENB*SWG01B, AND
3. Division ~~(3)~~ III, consisting of 125 volt D.C. distribution panel 1E22*S001PNL

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

~~One inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on the associated battery bank provided (1) its buses/MCCs/panels are OPERABLE and energized, and (2) the buses/MCCs/panels associated with the other battery banks are OPERABLE and energized.~~

SURVEILLANCE REQUIREMENTS (Continued)

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.

INSERT A →

3.4 By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses 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.

INSERT A

3. BY SELECTING AND FUNCTIONALLY TESTING A REPRESENTATNE SAMPLE OF AT LEAST 10% OF EACH TYPE OF MOTOR STARTER USED FOR PENETRATION REDUNDANT OVERCURRENT PROTECTION. MOTOR STARTERS SELECTED FOR FUNCTIONAL TESTING SHALL BE SELECTED ON A ROTATING BASIS. TESTING OF THESE MOTOR STARTERS SHALL CONSIST OF INJECTING A CURRENT WITH A VALUE EQUAL TO THE LOCKED ROTOR CURRENT OF THE ASSOCIATED MOTOR AND VERIFYING THAT THE MOTOR STARTER OPERATES TO INTERRUPT THE CURRENT WITHIN THE ASSOCIATED THERMAL OVERLOAD TIME DELAY BAND WIDTH FOR THAT CURRENT AS SPECIFIED BY THE MANUFACTURER. MOTOR STARTERS FOUND INOPERABLE DURING FUNCTIONAL TESTING SHALL BE RESTORED TO **OPERABLE** STATUS PRIOR TO RESUMING OPERATION. FOR EACH MOTOR STARTER FOUND INOPERABLE DURING THESE FUNCTIONAL TESTS, AN ADDITIONAL REPRESENTATIVE SAMPLE of AT LEAST 10% OF ALL THE MOTOR STARTERS OF THE INOPERABLE TYPE SHALL ALSO BE FUNCTIONALLY TESTED ULTIl NO MORE FAILURES ARE FOUND OR ALL MOTOR STARTERS OF THAT TYPE HAVE BEEN FUNCTIONALLY TESTED.

REFUELING OPERATIONS

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

(OLHOLD)

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. ~~One~~ Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to ~~23~~ feet above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish ~~SECONDARY CONTAINMENT INTEGRITY~~ Primary Containment within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 2 hours per 8-hour period.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each train consisting of at least:

a. One OPERABLE RHR pump, and

b. ~~One~~ Two OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than ~~(23)~~ feet above the top of the reactor pressure vessel flange.

ACTION:

a. With ~~less than~~ only one of the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode train.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

c. With no RHR shutdown cooling mode OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish ~~SECONDARY CONTAINMENT INTEGRITY~~ Primary Containment — Shutdown within 4 hours. added

SR 4.9.12 Duplication of 4.9.12.2, hence deleted 4.9.12.

SR 4.9.12.2 Typo in number, was 4.9.1.2.2, changed to 4.9.12.2.

LCO 3.9.12.e

SR 4.9.12.2.d Keylock switch at each terminal control. FSAR 9.1.4.2.3.11, p. 9.1-36.

LCO 3.9.12.f

SR 4.9.12.2.e Correct description,

LCO 3.9.12.g

SR 4.9.12.2.f FSAR 9.1.4.2.3.11, p. 9.1-37.

LCO 3.9.12.c

SR 4.9.12.2.b Deleted "Versa" because it was a vendor name and not appropriate to be in Tech Specs.

REFUELING OPERATIONS

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access doors and floor plugs of all rooms through which the transfer system penetrates are closed and locked.

b. All access ~~doors~~ interlocks and pedestal switches are OPERABLE.

c. The ~~Versa~~ blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. ~~All~~ At least one IFTS ~~primary and secondary~~ carriage position ~~and liquid level indicators are OPERABLE.~~ indicator shall be OPERABLE at each carriage position and at least one liquid level sensor shall be OPERABLE.

e. ~~The~~ All keylock switches which provides IFTS access control-transfer system lockout ~~is~~ are OPERABLE.

f. ~~All~~ The warning ~~flashing~~ lights outside of the access doors are OPERABLE.

APPLICABILITY: When the IFTS containment blank flange is removed.

ACTION:

With the requirements of the above specification not satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, verify that:

~~a. All access door interlocks are OPERABLE.~~ when the IFTS is in operation

~~b. The Versa blocking valve in the Fuel Building IFTS hydraulic power unit is OPERABLE.~~

a. c. All IFTS primary and secondary carriage position and level indicators are OPERABLE.

~~d. The keylock switch which provides IFTS access control-transfer system lockout is OPERABLE.~~

b e. ~~All~~ The warning ~~flashing~~ lights outside of the access doors ~~are~~ is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.9.12.1 Within 1 hour prior to the startup of the IFTS tube, verify that no personnel are in areas immediately adjacent to the ~~IFS~~ and that ~~all~~ the access doors to rooms through which the IFTS tube penetrates and floor plugs are closed and locked.

4.9.12.2 ~~4.9.1.2.2~~ Within 4 hours prior to the operation of IFTS and at least once per ~~12 hours~~ 7 days thereafter, verify that when the IFTS is in operation:

a. All access ~~door~~ interlocks and gate switches are OPERABLE.

b. The Versa blocking valve in the Fuel Handling Building IFTS hydraulic power unit is OPERABLE.

~~c. All IFTS primary and secondary carriage position and level indicators are OPERABLE.~~

c. ~~The~~ All keylock switches which provide IFTS access control-transfer system lockout ~~is~~ are OPERABLE.

~~d. All flashing lights outside of access doors are OPERABLE.~~

LCO 3.10.1 - "Primary" added to be consistent with Tech Spec 3/4.6.1.3 terminology.

LCO 3.10.1 & ACTION - 1% is an NRC requirement. GE-STS should be changed to delete parentheses. 1% is consistent with startup and refueling test requirements.

FSAR 14.2.10.1.6, p. 14.2-21; 14.2.12.3.4, p. 14.2-130, 131; 14.2.12.3.5, p. 14.2-132, 133, 134, 135; 14.2.12.3.6, p. 14.2-135, 136, 14.2.12.3.8, p. 14.2-138, 139.

DRAFT

RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases~~, from each reactor unit,~~ to areas at and beyond the SITE BOUNDARY (see Figure ~~3.1-3~~ 5.1.3-1) would exceed 0.3 mrem to any organ in a 31 day period.

(APPLICABILITY: At all times other than when the VENTILATION EXHAUST TREATMENT ~~system~~ SYSTEM is undergoing routine maintenance.)

ACTION:

a. With gaseous waste being discharged FROM THE VENTILATION EXHAUST DUCTS without treatment and in excess of the above limits, ~~in lieu of a Licensee Event Report,~~ prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days in accordance with the ODCM, METHODOLOGY AND PARAMETERS IN THE

Specification 3/4.11.2.6 provides the LCO and SR for Radioactive Gaseous Effluents as required by the Standard Radiological Effluent Technical Specification (RETS) for Boiling Water Reactors, to be implemented with the STS (NUREG-0123), Appendix A, Specifications.

The RETS and BWR's, NUREG-0473 (Rev. 2), is being revised to include editorial changes (permitted in applicable specifications) given in NUREG-0472 (Rev. 3) RETS for PWR's. Therefore the markup contains known edit changes adding the words, "whenever the main condenser evacuation system is in operation" to the requirement 4.11.2.6. RB-Unit 1 uses air ejectors for evacuation. The explanatory instructions in the title were deleted. At RB-Unit 1 the main condenser off gas treatment system is designed to withstand the effects of a hydrogen explosion and an alternative specification is not required. Ref. FSAR 11.3. The Alternative Specification, 3/4.11.2.6A, was deleted as not applicable.

Information used for the plant specified conditions was obtained from the River Bend FSAR Chapters 11.3. The Bases agrees with NUREG-0473 (Rev. 2) for systems designed to withstand a hydrogen explosion.

LCO 3.11.2.6

LCO, Action a, and SR 4.11.2.6 - Deleted oxygen. RB-U1 does not have O_2 monitors.

Action b - Deleted because it concerns instrumentation and is covered by Tech Spec 3/4.3.2.12, Table 3.3.7.12-1, Action 125.

Applicability - Changed for consistency with SR 4.11.2.6 and its Bases and Table 3.3.7.12-1, Footnote .

Surveillance Requirements 4.11.2.6 - See Tech Spec 3/4.3.7.12 - No oxygen monitors are provided or required so long as NUREG-0800, SRP11.3, is satisfied.

Replaced "Operable" with "in compliance with" to make action associated with hydrogen monitors in Tech Spec 3/4.3.7.12 Applicable.

TABLE ~~3.12-1~~ (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Food Products (cont'd)	Samples of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed (Ic10 - Ic13).	Monthly when available	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed (Ic20 - Ic23).	Monthly when available	Gamma isotopic[e] and I-131 analysis.

Footnote a - Changes per 10CFR50.73

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. ~~Prevent~~ Terminate inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

~~The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram.~~ The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken 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 measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The reactor protection system is made up of four logic channels. The logic channels A(A1) and C(A2) ~~to~~ comprise the one trip system and the logic channels B(B1) and D(B2) ~~to~~ comprise the other trip system ~~when complying~~ for determining compliance with technical specifications. Placement of either logic channel of a trip system in the tripped condition places the trip system in the tripped condition. The trip systems as defined above are independent of each other. There are usually four instrument channels (one in each logic channel) to monitor each parameter. The tripping of a logic channel in each trip system will result in a reactor scram.

Pages NA (B 3/4 3-2a, 3-1b, and 3-2b):

Deleted RBS does not have a solid state RPS

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring ~~capability~~ information instrumentation ensures that sufficient is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980).

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

~~3/4.3.7.8 CHLORINE (AND AMMONIA) DETECTION SYSTEM (Optional)~~

~~The OPERABILITY of the chlorine (and ammonia) detection system ensures that an accidental chlorine (and/or ammonia) release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine (and/or ammonia), the control room emergency ventilation system will automatically be placed in the (isolation) mode of operation to provide the required protection. (The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", (February 1975) (Revision 1, January, 1977).)~~

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	RT NDT °F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS

Replace This TABLE WITH THE ONE provided ON page B 3/4 4-6A

RI' ;END UNIT 1

BASES .E B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	P(%)	HIGHEST STARTING RT_{NDT} (^{o}F)	$RT^{MAX.*}_{NDT}$ (^{o}F)	AVG. UPPER SHELF (FT-LBS)	$RT^{MAX.*}_{NDT}$ EOL (^{o}F)
Plate	SA-533 GR B CL.1	C3138-2	0.08	0.012	+9	48	79	+57
Weld	SHELL COURSE NO.2 Vertical Seam 3	492L4871/ A421B27AF	0.03	0.020	-50	80	130	+30

NOTE:* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (^{o}F)
Shell Ring	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Flange	SA 508 Cl.2	ALL HEATS	+10
Vessel Flange	SA 508 Cl.2	ALL HEATS	+10
Feedwater Nozzle	SA 508 Cl.2	ALL HEATS	-20
Weld	LOW ALLOY STEEL	ALL HEATS	-20
Closure Studs	SA 540 GRADE B23 or B24	ALL HEATS	ISAE Section III, NB2300 Requirement of 45 Ft-lbs 25 mils lateral expansion at +10°F T.

Aug 13 1984

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCS system inoperable, subsystems of the RHR, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, ~~(a system for which no credit is taken in the safety analysis)~~, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system spray system and the LPCI subsystems of the RHR system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core ~~cooling systems~~ can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds ~~(100)~~ psig. This pressure is substantially below that for which the low pressure core ~~cooling systems~~ can provide adequate core cooling for events requiring ADS. (spray system and the LPCI subsystems of the RHR system)

ADS automatically controls (seven) selected safety-relief valves although the safety analysis only takes credit for (six) valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The supression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITIONS 4 and 5 the suppression ~~chamber~~ pool minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume and vortex prevention. ~~plus a (2'-4") safety margin for conservatism~~.

TECH SPEC 3/4.6.1.2 BASES - FSAR 6.2.1.1.1 (1.a), Page 6.2-2; 6.2.6.1, Page 6.2-89.

10CFR50 Appendix J - Appendix J, III D.2 (b) (III) changed since GE-STS corrected. Deleted airlock test exemption.

~~3/4.6.1 CONTAINMENT~~

3/4.6.1.1 PRIMARY CONTAINMENT ~~INTEGRITY~~

PRIMARY CONTAINMENT INTEGRITY and PRIMARY CONTAINMENT – SHUTDOWN

~~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 accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on PRIMARY containment leakage rates ensure that the total PRIMARY containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of ~~(15.0)~~ 6.31 psig, 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 tests to account for possible degradation of the PRIMARY containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing and ~~testing the airlocks after each opening.~~ ANSI/ANS 56.8-1981.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the PRIMARY containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the PRIMARY containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the PRIMARY containment.

TECH SPEC 3/4.6.1.4 BASES - RB-U1 design is a positive pressure system. MSIV leakage is not required to be measured or to be included in $0.60L_a$ limit. Added Bases for sealing air in-leakage limit.

TECH SPEC 3/4.6.1.5 BASES - FSAR 6.2.1.1.1 (1.a), Page 6.2-2; 6.2.6.1, Page 6.2-89.

TECH SPEC 3/4.6.1.6 BASES - Used RB-U1 terminology. FSAR 6.2.1.1.1 (1.a), Page 6.2-2; 6.2.6.1, Page 6.2-89

3/4.6.1.4 MSIV ~~LEAKAGE~~ POSITIVE LEAKAGE CONTROL SYSTEM

~~(Negative pressure systems)~~

~~Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines(, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact). Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIV's when isolation of the primary system and containment is required.~~

~~(Positive pressure systems)~~

~~Calculated doses resulting from the maximum leakage allowance for the isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines.~~ Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage control system will prevent untreated leakage from the MSIV's when isolation of the primary system and containment is required. THE MS-PLCS PREVENTS SUCH LEAKAGE.

3/4.6.1.5 ~~(Reinforced concrete containment)~~ PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the 6.31 PRIMARY containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the PRIMARY containment will withstand the maximum pressure of ~~45~~ psig in the event of a LOCA ~~(steam line break accident)~~. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations on PRIMARY containment ~~to secondary containment differential~~ INTERNAL pressure ensure that the PRIMARY containment peak pressure of ~~(12.0)~~ psig does not exceed the design pressure of 15.0 psig during LOCA ~~(steam line break)~~ conditions or that the external pressure differential does not exceed the design maximum external pressure differential of ~~(0.8)~~ psid for the differential at which water would overflow the wier wall into the drywell of 0.5 psid. The limit of ~~(0.1) to -(1.5) psid~~ for initial ~~positive~~ containment ~~to secondary containment~~ pressure will limit the containment pressure to ~~(12.0)~~ psig which is less than the design pressure and is consistent with the safety analysis. (-0.1" TO -0.1" WG).

3/4.6.1.7 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on PRIMARY containment average air temperature ensures that the PRIMARY containment peak air temperature does not exceed the design temperature of 185°F during LOCA ~~(steam line break)~~ conditions and is consistent with the safety analysis.

INSERT A

(LATER)

INSERT B

LEAKAGE OF SEALING AIR INTO THE PRIMARY CONTAINMENT IS LIMITED SUCH THAT, WHEN COMBINED WITH SEALING AIR INLEAKAGE FROM THE PVLCS, PRIMARY CONTAINMENT PRESSURE DOES NOT EXCEED 50% OF DESIGN VALUE AFTER 30 DAYS OF MS-PLCS AND PVLCS OPERATION.

INSERTS FOR
B 3/4 6-3

Aug 11, 1984

TECH SPEC 3/4.6.2.4 BASES - FSAR 6.2.6.5.1, Page 6.2-93.

TECH SPEC 3/4.6.2.5 BASES - FSAR 6.2.1.1.1 (1.a), p. 6.2-2;

BASES

3/4.7.1 ~~SERVICE~~ STANDBY SERVICE WATER SYSTEMS AND ULTIMATE HEAT SINK

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during ~~normal and~~ accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 ~~CONTROL~~ MAIN CONTROL ROOM EMERGENCY ~~FILTRATION~~ AIR CONDITIONING SYSTEM

The OPERABILITY of the ~~control~~ main control room ~~emergency filtration~~ air conditioning system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 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 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3. ~~FLOOD PROTECTION (Optional)~~

~~The requirement for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. The limit of elevation ( ) Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.~~

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds ~~(100)~~ 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds ~~(100)~~ 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

"And to start cooling at the earliest possible moment" deleted consistent with RCIC not being assumed in safety analysis.

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~~ demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers 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 than 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 OPERABILITY for bypassing of the motor operated valves thermal overload protection continuously or during accident conditions ~~(by integral bypass devices)~~ ensures that the thermal overload protection during accident conditions will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY or bypassing of the thermal overload protection continuously and/or 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.

B 3/4.9.3 Added "except removed control rods" to be consistent with * exception to LCO 3.9.3. The remainder is for clarity.

B 3/4.9.5 River Bend terminology consistent with markup of 3/4.9.5.

B 3/4.9.6 Added "/or" for clarity. No cranes are used for handling fuel assemblies or control rods in the reactor vessel.

DRAFT

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

5.1.3 INSERT HERE

5.2 CONTAINMENT

PRIMARY CONTAINMENT ~~CONFIGURATION~~

5.2.1 The primary containment is a steel torispherical ~~lined, reinforced concrete~~ structure composed of a vertical right cylinder and a ~~hemispherical~~ dome. Inside and at the bottom of the primary containment is a reinforced concrete drywell, composed of a vertical right cylinder and a steel head, which contains approximately 20 feet deep water filled suppression pool connected to the drywell through a series of horizontal vents. The primary containment has a minimum net free air volume of ~~(1,400,000)~~ cubic feet. The drywell has a minimum net free air volume of ~~(270,000)~~ 236,196 cubic feet. (1,191,590

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure: (25)
   1. Drywell ~~(30)~~ psig.
   2. Containment 15 psig.

b. Maximum internal temperature: (330)
   1. Drywell ~~(340)~~ °F.
   2. Suppression pool ~~(275)~~ (185) °F.

c. Maximum external to internal differential pressure:
   1. Drywell ~~(22)~~ psid. (20)
   2. Containment ~~(3)~~ (0.6) psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the ~~(Reactor~~ Shield Building, ~~the recirculation fan room, the equipment access structure and a portion of the main steam tunnel) and has a minimum free volume of (2,650,000) cubic feet.~~ the Auxilary Building and the Fuel Building. Secondary containment has a minimum combined free volume of 2,259,400 cubic feet.

~~d. type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),~~

~~e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and~~

~~f. solidification agent (e.g., cement, urea formaldehyde).~~

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

Insert 1 The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

Insert 2 MONTHLY REACTOR OPERATING REPORT (including documentation of all challenges to main steam safety / relief valves)

6.9.1.10 Routine reports of operating statistics, and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted ~~by the (Unit Review Group)~~.

Insert 3

~~PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP~~

~~6.9.1.12~~

~~j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.~~

~~k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.~~