FirstEnergy

DESIGN VERIFICATION RECORD

NOP-CC-2001-01 Rev. 00

SECTION I: TO BE COMPLETED BY DESIGN ORIGINATOR

DOCUMENT(S)/ACTIVITY TO BE VERIFIED:

Calc 3.2.15.14, Revision 0

☒ SAFETY RELATED ☐ AUGMENTED QUALITY ☐ NONSAFETY RELATED

SUPPORTING/REFERENCE DOCUMENTS

DESIGN ORIGINATOR: *(Print and Sign Name)* Hanry A. Wagage Hanry A. Wagage — DATE 1/3/02

SECTION II: TO BE COMPLETED BY VERIFIER

VERIFICATION METHOD *(Check one)*

☒ DESIGN REVIEW *(Complete Design Review Checklist or Calculation Review Checklist)* ☐ ALTERNATE CALCULATION ☐ QUALIFICATION TESTING

JUSTIFICATION FOR SUPERVISOR PERFORMING VERIFICATION:

The SCIENTECH manager performed this verification since this is an area for which he stays abreast of the industry positions and is known throughout the industry as an expert in this area.

APPROVAL: *(Print and Sign Name)* David A. Studley [signature] — DATE 1/4/02

EXTENT OF VERIFICATION:

The FHA calculation was completely reviewed, including the methodology, the use of the RADTRAD code, the conformance with current regulatory expectations, etc. This calculation was found to be acceptable and appropriate for inclusion with the LAR to the NRC. In addition, during the course of the calculation development, an analysis was done using an Excel spreadsheet which yielded identical results. Editorial comments and comments on the wording were provided to the preparer and have been incorporated.

COMMENTS, ERRORS OR DEFICIENCIES IDENTIFIED? ☒ YES ☐ NO

RESOLUTION: *(For Alternate Calculation or Qualification Testing only)*
Editorial comments provided and satisfactorily incorporated. Tech. comment noted on the check list regarding EDE for I-135 found acceptable as is.

RESOLVED BY: *(Print and Sign Name)* David A. Studley [signature] — DATE 1/4/02

VERIFIER: *(Print and Sign Name)* David A. Studley [signature] — DATE 1/4/02

APPROVED BY: *(Print and Sign Name)* M. Donovan M. Donovan — DATE 1/4/02

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
REFERENCES 1 Does the stated objective/purpose clearly describe why the calculation is being performed?		X			
2. Are applicable codes, standards, design/licensing basis documents, etc., including edition and addenda where appropriate clearly identified?		X			
3. Do the references reflect the appropriate revision?		X			
INPUTS 4. Are design inputs clearly identified and their source documents referenced, including revision level as appropriate?		X			
5. Are the design inputs relevant, current, consistent with design/licensing bases and directly applicable to the purpose of the calculation, including appropriate tolerances and ranges/modes of operation?		X			
6. Are all design inputs retrievable? If not, have they been added as attachments?		X			
7. Are preliminary or conceptual inputs clearly identified for later confirmation as open assumptions?	X				
ASSUMPTIONS 8. Have the assumptions necessary to perform the analysis been adequately documented?		X			
9. Is suitable justification provided for all assumptions (except those based upon recognized engineering practice, physical constants or elementary scientific principles)?		X			
10. Are all assumptions for the calculation reasonable and consistent with design/licensing bases?		X			
11. Have all open assumptions needing later confirmation been clearly identified on the Calculation cover sheet, including when the open assumption needs to be closed?	X				
12. Has a Condition Report been issued for open assumptions if required?		X			
13. Have engineering judgments been used?			X		
14. Are engineering judgments reasonable and adequately documented?	X				
METHOD OF ANALYSIS 15. Is the method used appropriate considering the purpose and type of calculation?		X			
16. Is the method in accordance with applicable codes, standards, and design/licensing bases?		X			
IDENTIFICATION OF COMPUTER CODES (Ref: NOP-SS-1001) 17. Have the versions of the computer codes employed in the design analysis been certified for this application?		X			
18. Are codes properly identified along with source, inputs and outputs?		X			
19. Is the code suitable for the analysis being performed?		X			
20. Does the computer model, that has been created, adequately reflect actual (or to be modified) plant conditions (e.g., dimensional accuracy, type of model/code options used, time steps, etc.)?		X			
21. Is the computer output reasonable when compared to inputs and what was expected?		X			
COMPUTATIONS 22. Are the equations used consistent with recognized engineering practice and design/licensing bases?		X			
23. Is justification provided for any equations not in common use?	X				
24. Is the justification reasonable?	X				
25. Have adjustment factors, uncertainties, empirical correlations, etc., used in the analysis been correctly applied?		X			
26. Is the result presented with proper units and tolerance?		X			
27. Has proper consideration been given to results that may be overly sensitive to very small changes in input?		X			

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
CONCLUSIONS					
28. Is the magnitude of the result reasonable when compared to inputs?		X			
29. Is the direction of trends reasonable?		X			
30. Are stated conclusions justifiable based on the calculation results?		X			
31. Are all pages sequentially numbered and marked with a valid calculation number?		X			
32. Is all information legible and reproducible?		X			
33. Have all changes in the documentation been initialed (or signed) and dated by the author of the change and all required reviewers?		X			
34. Have all calculation results stayed within existing design/licensing basis parameters?			X	This calc deviates from current design and licensing basis.	LAR being prepared to adopt this calculation for PNPP.
35. If the response to Question 34 is NO, has Licensing been notified as appropriate? (i.e. UFSAR or Tech Spec Change Request has been initiated).		X			
36. Does the calculation meet its purpose/objective?		X			
37. Has the calculation vendor used all applicable design information/requirements provided?		X			
38. Did the calculation vendor determine if the calculation was referenced in design basis documents and/or databases?		X			
39. Did the Preparer determine if the calculation was used as a reference in the UFSAR?		X			
40. If the calculation is used as a reference in the UFSAR, is a change to the UFSAR required or an update to the UFSAR Validation Database, if applicable, required?		X		USAR affected.	Once LAR is approved, the USAR will be updated. LAR includes USAR changes
41. If the answer to Question 40 is YES, have the appropriate documents been initiated?			X		See item 40.
42. Is the calculation acceptable for use?		X			
43. What checking method was used to review the calculation? Check all that apply.					
• spot check for math		X			
• complete check for math		X			
• comparison with tests					
• check by alternate method		X			
• comparison with previous calculation		X			

Review Summary: The FHA calculation, methodology, use of the code, etc. was completely reviewed and found acceptable. In addition, during the course of the calculation development, an analysis was done using an Excel spreadsheet which yielded identical results. Editorial comments provided to clarify the calculation. The EDE for I135 in the calc does not agree with FR12 (available at http://www.epa.gov/radiation/federal/docs/fgr12.pdf). The calc used a value from MACCS2 (DIN15) which is slightly higher than the FR12 value (conservative which is OK).

☒ **Technical Review**		☐ **Owner's Acceptance Review** *(Required for calculations prepared by a vendor)*	
Reviewer *(Print and Sign Name)* David A. Studley [signature]	Date 1/4/02	Reviewer: *(Print and Sign Name)* N/A	Date
		Approver: *(Print and Sign Name)* N/A	Date

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT Perry

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
REFERENCES 1 Does the stated objective/purpose clearly describe why the calculation is being performed?		x			
2. Are applicable codes, standards, design/licensing basis documents, etc., including edition and addenda where appropriate clearly identified?		x			
3. Do the references reflect the appropriate revision?		x			
INPUTS 4. Are design inputs clearly identified and their source documents referenced, including revision level as appropriate?		x			
5. Are the design inputs relevant, current, consistent with design/licensing bases and directly applicable to the purpose of the calculation, including appropriate tolerances and ranges/modes of operation?			x	Not necessarily consistent with design/license bases. License amendment request required.	N/A
6. Are all design inputs retrievable? If not, have they been added as attachments?		x			
7. Are preliminary or conceptual inputs clearly identified for later confirmation as open assumptions?	x				
ASSUMPTIONS 8. Have the assumptions necessary to perform the analysis been adequately documented?		x			
9. Is suitable justification provided for all assumptions (except those based upon recognized engineering practice, physical constants or elementary scientific principles)?		x			
10. Are all assumptions for the calculation reasonable and consistent with design/licensing bases?			x	See comment to item 5	N/A
11. Have all open assumptions needing later confirmation been clearly identified on the Calculation cover sheet, including when the open assumption needs to be closed?	X				
12. Has a Condition Report been issued for open assumptions if required?	X				
13. Have engineering judgments been used?		X			
14. Are engineering judgments reasonable and adequately documented?		X			
METHOD OF ANALYSIS 15. Is the method used appropriate considering the purpose and type of calculation?		X			
16. Is the method in accordance with applicable codes, standards, and design/licensing bases?			x	See comment to item 5	N/A
IDENTIFICATION OF COMPUTER CODES (Ref: NOP-SS-1001) 17. Have the versions of the computer codes employed in the design analysis been certified for this application?		x			
18. Are codes properly identified along with source, inputs and outputs?		x			
19. Is the code suitable for the analysis being performed?		x		Code is NRC endorsed	N/A
20. Does the computer model, that has been created, adequately reflect actual (or to be modified) plant conditions (e.g., dimensional accuracy, type of model/code options used, time steps, etc.)?		x			
21. Is the computer output reasonable when compared to inputs and what was expected?		x			
COMPUTATIONS 22. Are the equations used consistent with recognized engineering practice and design/licensing bases?			x	See comment to item 5	N/A
23. Is justification provided for any equations not in common use?	x				
24. Is the justification reasonable?	x				
25. Have adjustment factors, uncertainties, empirical correlations, etc., used in the analysis been correctly applied?		x			
26. Is the result presented with proper units and tolerance?		x			
27. Has proper consideration been given to results that may be overly sensitive to very small changes in input?		x		Refer to sensitivity analyses	N/A

FirstEnergy	**CALCULATION REVIEW CHECKLIST** NOP-CC-2001-04 Rev. 00	Page 2 of 2 CALCULATION NO. 3.2.15.14 REV. 0 UNIT Perry

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
CONCLUSIONS 28. Is the magnitude of the result reasonable when compared to inputs?		x			
29. Is the direction of trends reasonable?		x			
30. Are stated conclusions justifiable based on the calculation results?		x			
31. Are all pages sequentially numbered and marked with a valid calculation number?		x			
32. Is all information legible and reproducible?		x			
33. Have all changes in the documentation been initialed (or signed) and dated by the author of the change and all required reviewers?		x		Two typographical errors corrected as part of this review on page 16.	N/A
34. Have all calculation results stayed within existing design/licensing basis parameters?			x	See comment to item 5	N/A
35. If the response to Question 34 is NO, has Licensing been notified as appropriate? (i.e. UFSAR or Tech Spec Change Request has been initiated).		x		License amendment request has been prepared.	N/A
36. Does the calculation meet its purpose/objective?		x			
37. Has the calculation vendor used all applicable design information/requirements provided?		x			
38. Did the calculation vendor determine if the calculation was referenced in design basis documents and/or databases?	x			FE to prepare USAR Change and ATLAS database updates	N/A
39. Did the Preparer determine if the calculation was used as a reference in the UFSAR?		x			
40. If the calculation is used as a reference in the UFSAR, is a change to the UFSAR required or an update to the UFSAR Validation Database, if applicable, required?		x			
41. If the answer to Question 40 is YES, have the appropriate documents been initiated?		x		License amendment prepared.	N/A
42. Is the calculation acceptable for use?		x		Pending NRC approval	N/A
43. What checking method was used to review the calculation? Check all that apply.	X			Reviewed inputs to calculation.	N/A
• spot check for math	X			Did not re-run RADTRAD Code.	N/A
• complete check for math	X				
• comparison with tests	X				
• check by alternate method	X				
• comparison with previous calculation	X				

Review Summary: The USAR Change Request and ATLAS database update will consider the entire amendment request content and will be processed according to procedures upon NRC amendment approval. Owner's acceptance review looked at the calculation, sensitivities, appendix, as well as the attachments for approach, inputs, and conclusions. It is noted that Appendix A to the calculation evaluates a scenario that was identified in Condition Report 01-4224 as a result of a review of the draft amendment request. The associated amendment request was reviewed to ensure that the calculation and request were consistent.

☐ **Technical Review**		☒ **Owner's Acceptance Review** *(Required for calculations prepared by a vendor)*	
Reviewer *(Print and Sign Name)* N/A	Date	Reviewer: *(Print and Sign Name)* A. Widmer *A. Widmer*	Date 01-08-02
		Approver: *(Print and Sign Name)* THEODORE A WILSON *T.A. Wilson*	Date 01/08/02

6.5.2 Decay Time

OPERATIONAL REQUIREMENT: The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: MODE 5, during CORE ALTERATIONS.

ACTION: With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

Requirements remain unchanged

TESTING REQUIREMENTS:

6.5.2.1 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

BASES: The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

No changes are being made to this Operational Requirements Manual (ORM) page. Included for completeness.

INFORMATION ONLY

INFORMATION ONLY

(continued)

TABLE OF CONTENTS

INFORMATION ONLY

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Reactor Vessel Water Level—Low Low Low, Level 1 (continued)

This Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs. However, OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE.

This Function isolates the Group 2 isolation valves.

2.g. Containment and Drywell Purge Exhaust—Plenum Radiation—High

High purge exhaust plenum radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from four the primary containment due to a break in the RCPB. When Purge Exhaust-Plenum Radiation—High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products. ~~Additionally, the Purge Exhaust-Plenum Radiation—High is assumed to initiate isolation of the primary containment during a fuel handling accident involving handling of recently irradiated fuel (Ref. 2).~~ In addition, this Function provides an isolation signal to certain drywell isolation valves. The isolation of drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the drywell suppression function of the drywell.

The Purge Exhaust-Plenum Radiation—High signals are initiated from four radiation detectors that are located on the purge exhaust plenum ductwork coming from the drywell and containment. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.g Containment and Drywell Purge Exhaust-Plenum Radiation - High (continued)

Four channels of Containment and Drywell Purge Exhaust-Plenum Radiation-High Function are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Containment and Drywell Purge System inboard and outboard isolation valves each use a separate two-out-of-two isolation logic.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 100 limits (for the design-basis Revised Accident Source Term (RAST) LOCA analysis, the licensing basis offsite dose limit is 25 rem TEDE (Ref. 11)).

The Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs)~~, and movement of recently irradiated fuel assemblies in the primary containment~~ because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery ~~or dropped fuel assemblies~~) must be provided to ensure offsite dose limits are not exceeded. However, OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~

These Functions isolate the Group 8 valves.

2.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

(continued)

Enclosure 6
PY-CEI/NRR-2609L
Page 6 of 69

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.h Manual Initiation (continued)

Four channels of the Manual Initiation Function are required to be OPERABLE in MODES 1, 2, and 3, and during ~~movement of recently irradiated fuel assemblies in primary containment, or~~ operations with a potential for draining the reactor vessel, since these are the MODES in which the Primary Containment and Drywell Isolation automatic Functions are required to be OPERABLE. ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~ OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE.

The manual initiation channels for the RCIC System is discussed in Section 3.k below, and for the HPCS System is discussed in the Bases description for ECCS Instrumentation (LCO 3.3.5.1).

3. Reactor Core Isolation Cooling System Isolation

3.a. RCIC Steam Line Flow—High

RCIC Steam Line Flow—High Function is provided to detect a break of the RCIC steam lines and initiates closure of the steam line isolation valves. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and core uncovery can occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for this Function is not assumed in any USAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC steam line break from becoming bounding.

The RCIC Steam Line Flow—High signals are initiated from two transmitters that are connected to the system steam lines. Two channels of RCIC Steam Line Flow—High Functions are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

ACTIONS (continued)

K.1, ~~K.2.1~~ and K.2~~.2~~

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be isolated (Required Action K.1). Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable instrumentation. Alternately, the plant must be placed in a condition in which the LCO does not apply. ~~If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, i~~If applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission production release. Actions must continue until OPDRVs are suspended.

L.1

If applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Primary Containment and Drywell Isolation Instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains primary containment isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure-High (continued)

Drywell Pressure-High signals are initiated from four pressure transmitters that sense drywell pressure. Four channels of Drywell Pressure-High Function (two channels per trip system) are required to be OPERABLE to ensure that no single instrument failure can preclude CRER System initiation.

The Drywell Pressure-High Allowable Value was chosen to be the same as the ECCS Drywell Pressure-High Allowable Value (LCO 3.3.5.1).

The Drywell Pressure-High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. In MODES 4 and 5, the Drywell Pressure-High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure-High setpoint.

3. Control Room Ventilation Radiation Monitor

The Control Room Ventilation Radiation Monitor measures radiation levels downstream of the supply plenum discharge of the control room. A high radiation level may pose a threat to control room personnel; thus, the Control Room Ventilation Radiation Monitor Function will automatically initiate the CRER System.

The Control Room Ventilation Radiation Monitor Function consists of one noble gas monitor. One channel (which provides input to both Trip Systems) of the Control Room Ventilation Radiation Monitor is required to be OPERABLE. Since a LOCA signal will also initiate the CRER System isolating the control room from the environment, and considering the fact that a LOCA signal itself incorporates sufficient redundancy, the airborne radiation monitor signal is considered a diverse signal, and does not require redundancy. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Ventilation Radiation Monitor Function is required to be OPERABLE in MODES 1, 2, and 3, and during OPDRVs ~~and movement of recently irradiated fuel in the primary containment or Fuel Handling Building~~ to ensure

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Control Room Ventilation Radiation Monitor (continued)

that control room personnel are protected during a LOCA, ~~fuel handling event,~~ or a vessel draindown event. ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~ OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA or significant fuel damage is low; thus, the Function is not required.

ACTIONS

A Note has been provided to modify the ACTIONS related to CRER System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CRER System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CRER System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time an inoperable channel is discovered, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CRER System design, an allowable out of service time of 24 hours has been shown to be acceptable (Refs. 4 and 5) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable

(continued)

BASES

BACKGROUND (continued)

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.20% by weight of the containment and drywell air per 24 hours at the calculated maximum peak containment pressure (P_a) of 7.80 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the intermediate building.

The air locks are also required to be OPERABLE

During plant operations in other than MODES 1, 2, and 3, the primary containment contains the fission products from a Fuel Handling Accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) inside the primary containment (Ref. 4), to limit doses at the site boundary to within limits. The primary containment air lock OPERABILITY assures a leak tight fission product barrier during such activities.

i.e., during operations with the potential for draining the reactor vessel.

Primary containment air locks satisfy Criterion 3 and 4 (during shutdown) of the NRC Policy Statement.

LCO

As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air locks are required to be OPERABLE. For each air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock

(continued)

BASES

LCO (continued)

allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single OPERABLE door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from primary containment.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining OPERABLE primary containment air locks in MODE 4 or 5 to ensure a control volume is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs). ~~or during movement of recently irradiated fuel assemblies in the primary containment.~~ OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the primary containment air locks are not required to be OPERABLE. ~~Due to radioactive decay, handling of fuel only requires primary containment air lock OPERABILITY when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. However, even though the air locks are not required to be OPERABLE during handling of fuel that is not recently irradiated, there are still controls provided in an Operating License Condition to ensure the ability to close a door in an air lock should the need arise. Closure of a door, even though it is not OPERABLE, would reduce the potential for gross unfiltered leakage.~~

ACTIONS

The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. If the inner door is the one that is inoperable, then it is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time while operating in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1 ~~and E.2~~ 2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time during operations with a potential for draining the reactor vessel (OPDRVs). ~~or during movement of recently irradiated fuel assemblies in the primary containment,~~ 2 2 action is required to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, ~~movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ 2 2 2 2 2 action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program when in MODES 1, 2, and 3. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established prior to initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the combined Type B and C primary containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2.3

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure (Ref. 3), closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the nature of this interlock, and given that the interlock mechanism is only challenged when the primary containment air lock door is opened, this test is only required to be performed upon entering or exiting a primary containment air lock, but is not required more frequently than once per 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other administrative controls such as indications of air lock door status available to operations personnel.

SR 3.6.1.2.4

A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > 1.5 psig for a period of 24 hours from an initial pressure of 90 psig is an effective leakage rate test to verify system performance. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. USAR, Section 3.8.
2. 10 CFR 50, Appendix J, Option B.
3. USAR, Table 6.2-1.
4. ~~USAR, Section 15.7.6.~~
5. PNPP Safety Evaluation Report Supplement 7, Section 6.2.6 "Containment Leakage Testing," November 1985.

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs, are a loss of coolant accident (LOCA), and a main steam line break (MSLB). ~~and a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) inside primary containment (Refs. 1 and 2).~~ In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. It is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The inboard 42 inch purge supply and exhaust valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, and 3.

The outboard MSIVs must have a safety related air source available for use following an accident in order for leakage to be within limits. Therefore, anytime that this air source from the "B" train of P57 Safety Related Air System is not available, the outboard MSIVs may not be able to maintain valve leakage within the specified limits.

PCIVs satisfy Criterion 3 and 4 (during shutdown) of the NRC Policy Statement.

LCO

PCIVs form a part of the primary containment boundary and some also form a part of the RCPB. The PCIV safety function is related to minimizing the loss of reactor coolant inventory, and establishing primary containment boundary during a DBA.

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated

(continued)

LCO (continued)

The normally closed PCIVs or blind flanges are considered OPERABLE when, as applicable, manual valves are closed or opened in accordance with applicable administrative controls, automatic valves are de-activated and secured in their closed position, check valves with flow through the valve secured, or blind flanges are in place. The valves covered by this LCO with their associated stroke times, if applicable, are listed in Reference 3. Primary containment purge valves with resilient seals, secondary containment bypass valves, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment-Operating," as Type B or C testing.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory, and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE and the primary containment purge valves are not required to be sealed closed in MODES 4 and 5. Certain valves are required to be OPERABLE, however, to prevent inadvertent reactor vessel draindown and release of radioactive material. ~~during a postulated fuel handling accident involving handling of recently irradiated fuel.~~ These valves are those whose associated instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.) ~~Due to radioactive decay, handling of fuel only requires containment isolation valve OPERABILITY when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~

(continued)

INFORMATION ONLY

BASES

ACTIONS (continued)

F.1. ~~G.1~~, and F.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. If applicable, ~~movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable,~~ action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. If suspending the OPDRVs would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valves to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valves.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

Each inboard 42 inch (1M14-F045 and 1M14-F085) primary containment purge supply and exhaust isolation valve is required to be verified sealed closed at 31 day intervals because the primary containment purge valves are not fully qualified to close under accident conditions. This SR is designed to ensure that a gross breach of primary containment is not caused by an inadvertent opening of a primary containment purge valve. Detailed analysis of these purge supply and exhaust isolation valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Primary containment purge valves that are sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power, removing the air supply to the valve operator, or providing administrative control of the valve control switches. In this application, the term "sealed" has no connotation of leak tightness. The 31 day Frequency is based on primary containment purge valve use during unit operations.

This SR allows a valve that is open under administrative controls to not meet the SR during the time the valve is open. Opening a purge valve under administrative controls

(continued)

Enclosure 6
PY-CEI/NRR-2609L
Page 17 of 69

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.1 (continued)

is restricted to one valve in a penetration flow path at a given time (refer to discussion for Note 1 of the ACTIONS) in order to effect repairs to that valve. This allows one purge valve to be opened without resulting in a failure of the Surveillance and resultant entry into the ACTIONS for this purge valve, provided the stated restrictions are met. Condition D must be entered during this allowance, and the valve opened only as necessary for effecting repairs. Each purge valve in the penetration flow path may be alternately opened, provided one remains sealed closed, if necessary, to complete repairs on the penetration.

The SR is modified by a Note stating that the inboard 42 inch primary containment purge supply and exhaust isolation valves are only required to be sealed closed in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves and the subsequent release of radioactive material will exceed limits prior to the closing of the purge valves. At other times when the purge valves are required to be capable of closing (e.g., during ~~movement of recently irradiated fuel assemblies~~ OPDRVs), pressurization concerns are not present and the purge valves are allowed to be open.

SR 3.6.1.3.2

This SR verifies that the 18 inch (1M14-F190, 1M14-F195, 1M14-F200, and 1M14-F205) and outboard 42 inch (1M14-F040 and 1M14-F090) primary containment purge supply and exhaust isolation valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have purge valve leakage outside the limits (Condition D).

The SR is also modified by a Note (Note 1) stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. At times other than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during ~~movement of recently irradiated fuel assemblies~~ OPDRVs) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation capability would be required by SR 3.6.1.3.5, SR 3.6.1.3.7, and SR 3.6.1.3.8).

(continued)

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.5 (continued)

full closure isolation time is demonstrated by SR 3.6.1.3.7. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.1.3.6

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 4), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened. Additionally, a leak rate acceptance criteria of 0.05 L_a has been assigned to these valves.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during ~~handling of recently irradiated fuel~~ OPDRVs), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.7

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

demonstrated at the frequency of the leakage test requirements of the Primary Containment Leakage Rate Testing Program.

This SR is modified by a Note that states these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage rate limits are not applicable in these other MODES or conditions.

A second Note states that the Feedwater lines are excluded from this particular hydrostatic (water) testing program. This is because water leakage from the stem, bonnet and seat of the third, high integrity valves in the feedwater lines (the gate valves) is controlled by the Primary Coolant Sources Outside Containment Program (Technical Specification 5.5.2). The acceptance criteria for the Primary Coolant Sources Outside Containment Program is 7.5 gallons per hour.

SR 3.6.1.3.12

Verifying that each outboard 42 inch (1M14-F040 and 1M14-F090) primary containment purge supply and exhaust isolation valve is blocked to restrict opening to ≤ 50° is required to ensure that the valves can close under DBA conditions within the time limits assumed in the analyses of References 2 and 3.

The SR is modified by a Note stating that this SR is only required to be met in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when the purge valves are required to be capable of closing (e.g., during ~~movement of recently irradiated fuel assemblies in the primary containment~~ OPDRVs), pressurization concerns are not present, thus the purge valves can be fully open. The 24 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

(continued)

BACKGROUND (continued)

This Specification ensures that the performance of the primary containment, in the event of a ~~fuel handling accident involving handling of recently irradiated fuel, or~~ reactor vessel draindown, provides an acceptable leakage barrier to contain fission products, thereby minimizing offsite doses.

APPLICABLE SAFETY ANALYSES

During plant shutdown, The safety ~~design~~ basis for the primary containment is that it ~~contain the fission products from a fuel handling accident(s) involving handling of recently irradiated fuel inside the primary containment (Ref. 2)~~ provide a contained volume to limit fission product escape following an unanticipated water level excursion, to limit doses at the site boundary. ~~to within limits. The primary containment OPERABILITY in conjunction with the automatic closure of selected OPERABLE containment isolation valves (LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," and LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"), assures a leak tight fission product barrier. Its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage rates assumed in safety analyses.~~

The calculations for the fuel handling accident inside the primary containment do not credit the primary or secondary containment; ~~has been analyzed for two cases. In the first scenario, the fuel bundles involved are recently irradiated, i.e., they have occupied part of a critical reactor core within the previous seven days. The containment purge system is in operation and isolates on high radiation. This produces an immediate unfiltered release to the environment. The fission products which remain within the primary containment are conservatively assumed to be released at rates consistent with the DBA LOCA assumptions (e.g., 0.2% of the containment volume per day), and be filtered by the Annulus Exhaust Gas Treatment System prior to release to the environment.~~

~~In the second case, the fuel handling accident inside the primary containment is assumed to involve fuel bundles that have not been in a critical reactor core within the previous seven days. With the radioactive decay provided with this delay,~~ all gaseous fission products released from the water pool over the damaged fuel bundles are assumed to be immediately discharged directly to the environment (Ref. 2). Although there are no specific safety analyses for a reactor vessel draining event, the primary containment is required to be OPERABLE during OPDRVs.

Primary containment satisfies Criterion ~~3~~ 4 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Primary containment OPERABILITY is maintained by providing a contained volume to limit fission product escape following ~~a fuel handling accident involving handling of recently irradiated fuel, or~~ an unanticipated water level excursion. Compliance with this LCO will ensure a primary containment configuration, including the equipment

(continued)

LCO (continued)

hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Since offsite dose analyses conservatively assume LOCA leakage pathways and rates, the isolation and closure times of automatic containment isolation valves supports an OPERABLE primary containment during shutdown conditions. Furthermore, normal operation of the inclined fuel transfer system (IFTS) without the IFTS blind flange installed is considered acceptable for meeting Primary Containment-Shutdown OPERABILITY.

Leakage rates specified for the primary containment and air locks, addressed in LCO 3.6.1.1 and LCO 3.6.1.2 are not directly applicable during the shutdown conditions addressed in this LCO.

APPLICABILITY

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining an OPERABLE primary containment in MODE 4 or 5 to ensure a control volume, is only required during situations for which significant releases of radioactive material can be postulated; such as during ~~movement of recently irradiated fuel assemblies in the primary containment, or during~~ operations with a potential for draining the reactor vessel (OPDRVs). ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of Primary Containment when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~ OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the primary containment is not required to be OPERABLE.

ACTIONS

A.1 ~~and A.2~~

In the event that primary containment is inoperable, action is required to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, ~~movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ action must be

(continued)

ACTIONS

A.1 and A.2 (continued)

immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.10.1

This SR verifies that each primary containment penetration that could communicate gaseous fission products to the environment during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed manual valve, a closed and de-activated automatic valve, and a blind flange. This SR does not require any testing or isolation device manipulation. Rather, it involves verification that these isolation devices capable of being mispositioned are in the correct position. The 31 day Frequency was chosen to provide added assurance that the isolation devices remain in the correct positions.

This SR is modified by three Notes. The first Note does not require this SR to be met for pathways capable of being isolated by OPERABLE primary containment automatic isolation valves. The second Note permits the Fire Protection System manual hose reel containment isolation valves (1P54-F726 and 1P54-F727) to be open during shutdown conditions to supply fire mains. The third Note is included to clarify that manual valves opened under administrative controls are not required to meet the SR during the time the manual valves are open.

REFERENCES

1. Deleted.
2. USAR, Section 15.7.6.

No changes to this page; provided for information

APPLICABLE SAFETY ANALYSES (continued)

b. Inadvertent actuation of both primary RHR containment spray subsystems during normal operation;

The results of these two cases show that the containment vacuum breakers, with an opening setpoint of 0.1 psid, are capable of maintaining the differential pressure within design limits.

The containment vacuum breakers satisfy Criterion 3 and 4 (during shutdown) of the NRC Policy Statement.

LCO

Only 3 of the 4 vacuum breakers must be OPERABLE for opening. All containment vacuum breakers, however, are required to be closed (except during testing or when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the containment negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, the RHR Containment Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside the containment could occur due to inadvertent actuation of this system. The vacuum breakers, therefore, are required to be OPERABLE in MODES 1, 2, and 3, to mitigate the effects of inadvertent actuation of the RHR Containment Spray System.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining containment vacuum breakers OPERABLE is not required in MODE 4 or 5.

~~When handling recently irradiated fuel in the primary containment, and~~ During operations with a potential for draining the reactor vessel (OPDRVs) the primary containment is required to be OPERABLE. Containment vacuum breakers are therefore required to be OPERABLE during these evolutions to protect the primary containment against an inadvertent initiation of the Containment Spray System. ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of Containment Vacuum Breakers when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~ Since OPDRVs assume that one or more fuel assemblies are loaded into the core, this LCO would not be applicable for OPDRVs if no fuel is in the reactor vessel.

(continued)

ACTIONS

A.1 and A.2 (continued)

A Note has been added to provide clarification that separate Condition entry is allowed for each containment vacuum breaker.

B.1 and B.2

If the Required Action of Condition A cannot be met, or if there are three or more containment vacuum breakers not closed, or if there are two or three required vacuum breakers inoperable for other reasons, the plant must be brought to a MODE or condition in which the LCO does not apply. To achieve this status, if the plant is operating, ACTION B.1 requires that the plant be brought to at least MODE 3 within 12 hours and that the plant be brought to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. A Note has been added to stipulate that these Required Actions are only applicable if the plant is in MODE 1, 2, or 3.

If the Condition occurs during ~~movement of recently irradiated fuel in the primary containment, or during~~ operations with a potential for draining the reactor vessel (OPDRVs), then ACTION B.2 requires that action be taken to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, ~~movement of recently irradiated fuel in the primary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ action must be taken to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended. A Note has been added to the Required Actions to stipulate that these requirements are only applicable ~~while moving recently irradiated fuel assemblies in the primary containment, or~~ during OPDRVs.

(continued)

LCO (continued)

relative humidity are not required to be maintained within the prescribed limits.

APPLICABILITY

In MODES 1, 2, and 3, the RHR Containment Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside the containment could occur due to inadvertent actuation of this system. The containment average temperature relationship with relative humidity, therefore, is required to be within limits in MODES 1, 2, and 3, to mitigate the effects of inadvertent actuation of the RHR Containment Spray System.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES. Therefore, maintaining limits on containment relative humidity and temperature is not required in MODE 4 or 5.

~~When handling recently irradiated fuel in the primary containment, and~~ During operations with a potential for draining the reactor vessel (OPDRVs) the primary containment is required to be OPERABLE. Therefore, the proper relationship between containment average temperature and relative humidity must exist during these evolutions. ~~Due to radioactive decay, handling of fuel only requires control over Containment humidity when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~

ACTIONS

A.1

With the primary containment average temperature and relative humidity not within the established limits, actions must be taken to restore the primary containment relative humidity and temperature to within limits. Required Action A.1 stipulates that restoration must occur within 8 hours. The eight hour Completion Time is based on the time required to restore the relative humidity and temperature limits, and the low probability of an event occurring during this time period.

(continued)

ACTIONS (continued)

B.1 and B.2

If the primary containment relative humidity and temperature cannot be restored to within limits within the required Completion Time of Condition A, actions must be taken to place the plant in a MODE or condition in which the LCO does not apply.

Required Action B.1 requires that the plant be brought to at least MODE 3 within 12 hours and Required Action B.2 requires that the plant be brought to MODE 4 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 ~~and C.2~~

If the primary containment relative humidity and temperature cannot be restored to within limits within the required completion time of Condition A during ~~movement of recently irradiated fuel in the primary containment, or during~~ OPDRVs, action is required to place the plant in a MODE or condition in which the LCO does not apply.

Required Actions C.1 and C.2 require that actions be taken to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a condition that minimizes risk.

If applicable, ~~movement of recently irradiated fuel in the primary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ actions must be taken to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

INFORMATION ONLY

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Annulus Exhaust Gas Treatment (AEGT) System and manual closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, such as during ~~movement of recently irradiated fuel assemblies in the primary containment, or during~~ operations with a potential for draining the reactor vessel (OPDRVs).

The secondary containment is a structure that completely encloses the primary containment. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the external pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Annulus Exhaust Gas Treatment (AEGT) System."

The isolation devices for the penetrations in the secondary containment boundary are a part of the secondary containment barrier. To maintain this barrier:

a. All penetrations terminating in the secondary containment required to be closed during accident conditions are closed by at least one manual valve or blind flange, as applicable, secured in its closed position, except as provided in LCO 3.6.4.2, Secondary Containment Isolation Valves (SCIVs)";

(continued)

BACKGROUND (continued)

b. The containment equipment hatch is closed and sealed and the shield blocks are installed adjacent to the shield building;

c. The door in each access to the secondary containment is closed, except for entry and exit;

d. The sealing mechanism associated with each shield building penetration, e.g. welds, bellows, or O-rings, is functional;

e. The pressure within the secondary containment is less than or equal to the value required by Surveillance Requirement SR 3.6.4.1.1, except for entry and exit to the annulus; and

f. The Annulus Exhaust Gas Treatment System is OPERABLE.

APPLICABLE SAFETY ANALYSES

There ~~are two~~ is one principal accident~~s~~ for which credit is taken for secondary containment OPERABILITY. ~~These are a~~ This is the LOCA (Ref. 1) ~~and a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) inside primary containment (Ref. 2).~~ The secondary containment performs no active function in response to ~~each of these~~ the limiting event~~s~~; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the AEGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 and 4 (during shutdown) of the NRC Policy Statement.

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

(continued)

BASES (continued) INFORMATION ONLY

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during ~~movement of recently irradiated fuel assemblies in the primary containment, or during~~ operations with a potential for draining the reactor vessel (OPRDVs). ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of Secondary Containment when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days~~. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the secondary containment is not required to be OPERABLE.

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

ACTIONS (continued)

C.1 ~~and C.2~~

~~Movement of recently irradiated fuel assemblies in the primary containment and~~ OPDRVs can be postulated to cause significant fission product releases. In such ~~cases~~ case, the secondary containment is one of the barriers to release of fission products to the environment. If applicable, ~~movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended if the secondary containment is inoperable. Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, if applicable,~~ action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring between surveillances.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that the primary containment equipment hatch is closed and the shield blocks are installed adjacent to the shield building, and secondary containment access doors are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. In this application, the term "sealed" has no connotation of leak tightness. Verifying that all such openings are closed provides adequate

(continued)

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.3 (continued)

assurance that exfiltration from the secondary containment will not occur. Maintaining secondary containment OPERABILITY requires verifying each door in both access openings are closed, except when the access opening is being used for entry and exit. The 31 day Frequency for these SRs has been shown to be adequate based on operating experience, and is considered adequate in view of the other controls on secondary containment access openings.

REFERENCES

1. USAR, Section 15.6.5.

~~2. USAR, Section 15.7.6.~~

96-109

Enclosure 6
PY-CEI/NRR-2609L
Page 33 of 69

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1).

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Isolation barrier(s) for the penetration are discussed in Reference 2. The isolation devices addressed by this LCO are passive. Manual valves and blind flanges are considered passive devices.

Penetrations are isolated by the use of manual valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accident~~s~~ for which the secondary containment boundary is required ~~are~~ is a loss of coolant accident (Ref. 1). ~~and a fuel handling accident involving handling of recently irradiated fuel~~ (i.e., fuel that has ~~occupied part~~ of a critical reactor core within the ~~previous~~ seven days) inside primary containment (Ref. 3). The secondary containment performs no active function in response to ~~each of these~~ this limiting event~~s~~, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Annulus Exhaust Gas Treatment (AEGT) System before being released to the environment.

Maintaining SCIVs OPERABLE ensures that fission products will remain trapped inside secondary containment so that they can be treated by the AEGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 and 4 (during shutdown) of the NRC Policy Statement.

(continue)

APPLICABLE SAFETY ANALYSES

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed, or open in accordance with appropriate administrative controls, or blind flanges are in place. The valves covered by this LCO are included in Table B 3.6.4.2-1.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during ~~movement of recently irradiated fuel assemblies in the primary containment, or during~~ operations with a potential for draining the reactor vessel (OPDRVs). ~~Due to radioactive decay, handling of fuel only requires OPERABILITY of secondary containment isolation valves when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~ OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the SCIVs are not required to be OPERABLE.

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when the need for secondary containment isolation is indicated.

The second Note provides clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

INFORMATION ONLY

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With two SCIVs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed manual valve, and a blind flange. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the low probability of a DBA occurring during this short time.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 ~~and D.2~~

If any Required Action and associated Completion Time of Condition A or B cannot be met during ~~movement of recently irradiated fuel assemblies in the primary containment.~~

(continued)

ACTIONS

D.1 and D.2 (continued)

~~or during~~ OPDRVs, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment isolation manual valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or isolation device manipulation. Rather, it involves verification that those isolation device in secondary containment that are capable of being mispositioned are in the correct position.

Since these isolation devices are readily accessible to personnel during normal unit operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the isolation devices are in the correct positions.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices once they have been verified to be in the proper position, is low. A second Note has been included to clarify that

(continued)

INFORMATION ONLY

Enclosure 6
PY-CEI/NRR-2609L
Page 37 of 69

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1 (continued)

that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open.

REFERENCES

1. USAR, Section 15.6.5.
2. USAR, Section 6.2.3.
3. ~~USAR, Section 15.7.6.~~

96-109

BACKGROUND (continued)

humidity of the airstream to less than 70% (Ref. 2). The roughing filter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The AEGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. AEGT System flows are controlled by two motor operated control dampers installed in branch ducts. One duct exhausts air to the unit vent, (AEGT Subsystem A exhausts to the Unit 1 plant vent; AEGT Subsystem B exhausts to the Unit 2 plant vent), while the other recirculates air back to the annulus.

APPLICABLE SAFETY ANALYSES

The design basis for the AEGT System is to mitigate the consequences of a loss of coolant acciden~~t and fuel handling accidents involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days (Ref. 2)~~. For all events analyzed, the AEGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The AEGT System satisfies Criterion 3 and 4 (during shutdown) of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one AEGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two independent operable subsystems ensures operation of at least one AEGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, AEGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the AEGT System OPERABLE is not required in MODE 4 or 5, except for

(continued)

APPLICABILITY (continued)

other situations under which significant releases of radioactive material can be postulated, such as during ~~movement of recently irradiated fuel assemblies in the primary containment, or during~~ operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of the AEGT System when the fuel ~~being~~ handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the AEGT System is not required to be OPERABLE.

ACTIONS

A.1

With one AEGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE AEGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant AEGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the AEGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, ~~C.2.1 and~~ C.2.2

~~During movement of recently irradiated fuel assemblies in the primary containment, or~~ During OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE AEGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no

(continued)

ACTIONS

C.1, ~~C.2.1 and~~ C.2.2 (continued)

failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, ~~movement of recently irradiated fuel~~ assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

D.1

If both AEGT subsystems are inoperable in MODE 1, 2, or 3, the AEGT System may not be capable of supporting the required radioactivity release control function. Therefore, LCO 3.0.3 must be entered immediately.

E.1 ~~and E.2~~

When two AEGT subsystems are inoperable, if applicable, ~~movement~~ of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each AEGT subsystem from the control room for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA

(continued)

INFORMATION ONLY

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRER System OPERABLE is not required in MODE 4 or 5, except ~~for the following situations under which significant radioactive releases can be postulated:~~

~~a. During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building; and~~

~~b.~~ during operations with a potential for draining the reactor vessel (OPDRVs).

~~Due to radioactive decay, handling of fuel only requires OPERABILITY of the Control Room Emergency Recirculation System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~ OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the CRER System is not required to be OPERABLE.

ACTIONS

A.1

With one CRER subsystem inoperable, the inoperable CRER subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRER subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE CRER subsystem could result in loss of CRER System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining CRER subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CRER subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

INFORMATION ONLY

ACTIONS (continued)

C.1, ~~C.2.1~~ and C.2 2

~~The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown. During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or~~ During OPDRVs, if the inoperable CRER subsystem cannot be restored to OPERABLE status within the required Completion Time of Condition A, the OPERABLE CRER subsystem may be placed in the emergency recirculation mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

~~If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also,~~ If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both CRER subsystems are inoperable in MODE 1, 2, or 3, the CRER System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

ACTIONS (continued)

E.1 ~~and E.2~~

~~During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or~~ During OPDRVs, with two CRER subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

~~If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also~~ If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Operating each CRER subsystem for ≥ 10 continuous hours after initiating from the control room and ensuring flow through the HEPA filters and charcoal adsorbers ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.7.3.2

This SR verifies that the required CRER testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRER filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter efficiency, charcoal adsorber efficiency and bypass leakage, system flow rate, and general operating

(continued)

INFORMATION ONLY

LCO

Two independent and redundant subsystems of the Control Room HVAC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room HVAC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers with compressors, ductwork, dampers, and associated instrumentation and controls. The heating coils are not required for control room HVAC OPERABILITY.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room HVAC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room HVAC System OPERABLE is not required in MODE 4 or 5, except ~~for the following situations under which significant radioactive releases can be postulated~~:

~~a. During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building; and~~

~~b.~~ during operations with a potential for draining the reactor vessel (OPRDVs).

~~Due to radioactive decay, handling of fuel only requires OPERABILITY of the Control Room HVAC System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~

OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the Control Room HVAC System is not required to be OPERABLE.

(continued)

Enclosure 6
PY-CEI/NRR-2609L
Page 45 of 69

ACTIONS (continued)

D.1, ~~D.2.1~~, and D.2

~~The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

~~During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or~~ During OPDRVs, if the inoperable control room HVAC subsystem cannot be restored to OPERABLE status within the required Completion Time of Condition A, the OPERABLE control room HVAC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

~~If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also,~~ If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

ACTIONS (continued)

E.1 ~~and E.2~~

~~The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

~~During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or~~ During OPDRVs if the Required Action and associated Completion Time of Condition B is not met, action must be taken to immediately suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

~~If applicable, handling of recently irradiated fuel in the primary containment or fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also,~~ If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analysis. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room HVAC System is not expected over this time period.

REFERENCES

1. USAR, Section 6.4.
2. USAR, Section 9.4.1.

B 3.7 PLANT SYSTEMS

Enclosure 6
PY-CEI/NRR-2609L
Page 47 of 69

B 3.7.7 Fuel Pool Water Level

INFORMATION ONLY

BASES

BACKGROUND

The minimum water level in the spent fuel storage pools and upper containment fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the fuel handling building (FHB) spent fuel storage pools and upper containment fuel storage pool design is found in the USAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Sections 15.7.4 and 15.7.6 (Refs. 2 and 3, respectively).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the offsite radiological consequences (~~calculated whole body and thyroid~~ Total Effective Dose Equivalent (TEDE) doses at the exclusion area and low population zone boundaries) are ≤ 25% ~~(NUREG-0800, Section 15.7.4, Ref. 4)~~ of the 10 CFR ~~100~~ 50.67 (Ref. 5) exposure guidelines. The control room is also evaluated to ensure doses are less than the 10 CFR 50.67 exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.~~25~~ 183 (Ref. ~~6~~ 7).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto stored fuel bundles. The consequences of a fuel handling accident inside the FHB and inside containment are documented in References 2 and 3, respectively. The water levels in the FHB spent fuel storage pools and upper containment fuel storage pools provide for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The fuel pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

No changes to this page, provided for completeness

LCO

The specified water level preserves the assumption of the fuel handling accident analysis (Refs. 2 and 3). As such, it is the minimum required for fuel movement within the FHB spent fuel storage pools and upper containment fuel storage pool.

APPLICABILITY

This LCO applies whenever movement of irradiated fuel assemblies occurs in the associated fuel storage racks since the potential for a release of fission products exists.

ACTIONS

A.1

INFORMATION ONLY

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. With either fuel storage pool level less than required, the movement of irradiated fuel assemblies in the associated fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the FHB spent fuel storage pools and upper containment fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pools is normally stable and water level changes are controlled by unit procedures.

(continued)

INFORMATION ONLY

REFERENCES

1. USAR, Section 9.1.2.
2. USAR, Section 15.7.4.
3. USAR, Section 15.7.6.
4. ~~NUREG 0800, Section 15.7.4, Revision 1, July 1981.~~ Deleted
5. 10 CFR ~~100.~~ 50.67
6. ~~Regulatory Guide 1.25, March 1972.~~ Deleted
7. Regulatory Guide 1.183, July 2000.

B 3.7 PLANT SYSTEMS

INFORMATION ONLY

B 3.7.8 Fuel Handling Building (FHB)

BASES

All the Bases for this Spec are removed

BACKGROUND

The function of the FHB is to contain, dilute, and hold up fission products that are released from a design basis Fuel Handling Accident (FHA). In conjunction with operation of the FHB Ventilation Exhaust System, the FHB is designed to reduce the activity level of the fission products prior to release to the environment.

The FHB is a three story building, located between the Unit 1 and 2 reactor buildings. The entire exterior of the building is reinforced concrete, including the foundation, the walls, and the roof slabs. The FHB houses four pools for fuel handling and storage:

a. Cask pit;

b. Spent fuel storage pool;

c. Fuel transfer pool; and

d. Fuel storage and preparation pool.

The pools are interconnected, by means of gates, to allow the underwater passage of fuel assemblies from one pool to another.

To prevent ground level exfiltration, the FHB boundaries have been established. The FHB boundaries are:

a. The doors in each access to the 620 foot elevation of the FHB are closed, except for normal entry and exit;

b. The FHB railroad track door is closed;

c. The FHB floor hatches are in place; and

d. The shield blocks are installed adjacent to the shield building.

(continued)

INFORMATION ONLY

Enclosure 6
PY-CEI/NRR-2609L
Page 51 of 69

B 3.7 PLANT SYSTEMS

B 3.7.9 Fuel Handling Building (FHB) Ventilation Exhaust System

BASES

All the Bases for this Spec are removed

BACKGROUND

The FHB Ventilation Exhaust System is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1). The function of the FHB Ventilation Exhaust System is to ensure that radioactive materials that escape from fuel assemblies damaged following a design basis Fuel Handling Accident (FHA) are filtered and adsorbed prior to exhausting to the environment.

The FHB Ventilation Exhaust System consists of three fully redundant subsystems, each with its own set of ductwork, dampers, exhaust fan, charcoal filter train, instrumentation, and controls. Each subsystem is designed for 50% flow.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

a. A demister;

b. A roughing filter;

c. Electric heater coils;

d. A high efficiency particulate air (HEPA) prefilter;

e. A charcoal adsorber; and

f. A HEPA afterfilter.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The roughing filter removes large particulate matter, while the HEPA prefilter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the HEPA afterfilter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The FHB Ventilation Exhaust System is manually started by the control room operators. In the event that the radiation monitor upstream of the charcoal filter trains senses a high

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources – Shutdown

INFORMATION ONLY

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 4 and 5 ~~and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building~~ ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel ~~or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~

In general, when the unit is shut down the Technical Specifications (TS) requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence significantly reduced or eliminated, and minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for required systems.

(continued)

INFORMATION ONLY

BASES

LCO (continued)

powered from offsite power. An OPERABLE DG, associated with a Division 1 or Division 2 Distribution System Engineered Safety Feature (ESF) bus required OPERABLE by LCO 3.8.8, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Similarly, when the high pressure core spray (HPCS) system is required to be OPERABLE, a separate offsite circuit to the Division 3 Class 1E onsite electrical power distribution subsystem, or an OPERABLE Division 3 DG, ensure an additional source of power for the HPCS. This additional source for Division 3 is not necessarily required to be connected to be OPERABLE. Either the circuit required by LCO Item a, or a circuit required to meet LCO Item c may be connected, with the second source available for connection. Together, OPERABILITY of the required offsite circuit(s) and DG(s) ensure the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., ~~fuel handling accidents involving handling of recently irradiated fuel,~~ reactor vessel draindown).

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective ESF bus(es), and accepting required loads during an accident. Qualified offsite circuits are those that are described in the USAR and are part of the licensing basis for the plant. One offsite circuit consists of the Unit 1 startup transformer through the Unit 1 interbus transformer, to the Class 1E 4.16 kV ESF buses through source feeder breakers for each required division. A second acceptable offsite circuit consists of the Unit 2 startup transformer through the Unit 2 interbus transformer, to the Class 1E 4.16 kV ESF buses through source feeder breakers for each required division.

The required DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 10 seconds for Division 1 and 2 and 13 seconds for Division 3. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot and DG in standby

(continued)

LCO (continued)

with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY. In addition, proper load sequence operation is an integral part of offsite circuit and DG OPERABILITY since its inoperability impacts the ability to start and maintain energized loads required OPERABLE by LCO 3.8.8. It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required AC electrical power distribution subsystems.

As described in Applicable Safety Analyses, in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 4 and 5 ~~and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building~~ provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

~~b. Systems needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the AC Sources when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days);~~

b ~~c~~. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c ~~d~~. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

(continued)

INFORMATION ONLY

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1

A required offsite circuit is considered inoperable if no qualified circuit is supplying power to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.8, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, ~~movement of recently irradiated fuel,~~ and operations with a potential for draining the reactor vessel.

By allowing the option to declare required features inoperable which are not powered from offsite power, appropriate restrictions can be implemented in accordance with the required feature(s) LCOs' ACTIONS. Required features remaining powered from offsite power (even though that circuit may be inoperable due to failing to power other features) are not declared inoperable by this Required Action.

A.2.1, A.2.2, A.2.3, ~~A.2.4,~~ B.1, B.2, B.3, and ~~B.4~~

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, ~~movement of recently irradiated fuel assemblies in the primary containment and fuel handling building,~~ and operations with a potential for draining the reactor vessel. Additionally, crane operations over the spent fuel storage pool shall be suspended when fuel assemblies are stored there.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to initiate

(continued)

Enclosure 6
PY-CEI/NRR-2609L
Page 56 of 69

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources – Shutdown

INFORMATION ONLY

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 ~~and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building~~ ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel. ~~or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.~~

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One DC electrical power subsystem (consisting of either the Unit 1 or 2 battery, either the normal or reserve battery charger, and all the associated control equipment and interconnecting cabling supplying power to the associated

(continued)

LCO (continued)

bus), associated with the Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem(s) required OPERABLE by LCO 3.8.8, "Distribution Systems-Shutdown," is required to be OPERABLE. Similarly, when the High Pressure Core Spray (HPCS) System is required to be OPERABLE, the Division 3 DC electrical power subsystem associated with the Division 3 onsite Class 1E DC electrical power distribution subsystem required OPERABLE by LCO 3.8.8 is required to be OPERABLE. In addition to the preceding subsystems required to be OPERABLE, a Class 1E battery or battery charger and the associated control equipment and interconnecting cabling capable of supplying power to the remaining Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem, when portions of both Division 1 and Division 2 DC electrical power distribution subsystems are required to be OPERABLE by LCO 3.8.8. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., ~~fuel handling accidents involving handling of recently irradiated fuel and~~ inadvertent reactor vessel draindown).

Division 1 consists of :
1. 125 volt battery 1R42-S002 or 2R42-S002.
2. 125 volt full capacity charger 1R42-S006 or OR42-S007.

Division 2 consists of:
1. 125 volt battery 1R42-S003 or 2R42-S003.
2. 125 volt full capacity charger 1R42-S008 or OR42-S009.

Division 3 consists of:
1. 125 volt battery 1E22-S005 or 2E22-S005.
2. 125 volt full capacity charger 1E22-S006 or OR42-S011.

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 ~~and during movement of recently irradiated fuel assemblies in the primary containment and fuel handling building~~ provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

(continued)

APPLICABILITY (continued)

b. Required features needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the DC Sources when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days);

(continued)

INFORMATION ONLY

Enclosure 6
PY-CEI/NRR-2609L
Page 59 of 69

BASES

APPLICABILITY (continued)

b ~~c~~. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. ~~d~~. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and ~~A.2.4~~ e

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, ~~movement of recently irradiated fuel,~~ e and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable DC power source(s) inoperable, appropriate restrictions are implemented in accordance with the Required Actions of the LCOs for these associated required features. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative alternate actions (i.e., to suspend CORE ALTERATIONS, ~~movement of recently irradiated fuel assemblies in the primary containment and fuel handling building,~~ e and operations with a potential for draining of the reactor vessel) is made.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems – Shutdown

INFORMATION ONLY

BASES

BACKGROUND

A description of the AC and DC electrical power distribution systems is provided in the Bases for LCO 3.8.7, "Distribution Systems – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC and DC electrical power distribution systems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 ~~and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building~~ ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling ~~accident~~ involving handling of recently ~~irradiated~~ fuel, i.e., fuel that has ~~occupied~~ part of a critical reactor core ~~within the~~ previous seven days.

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

INFORMATION ONLY

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the AC and DC electrical power distribution systems necessary to support OPERABILITY of Technical Specifications' required systems, equipment, and components – both specifically addressed by their own LCOs, and implicitly required by the definition of OPERABILITY.

Maintaining these portions of the AC and DC electrical power distribution systems energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., ~~fuel handling accidents involving handling of recently irradiated fuel and~~ inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 ~~and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building~~ provide assurance that:

a. Required features needed to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

~~b. Required features needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the Distribution Systems when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days);~~

b ~~c~~. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

(continued)

INFORMATION ONLY

BASES

APPLICABILITY (continued)

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

(continued)

 99-027

ACTIONS

The ACTIONS are modified by a Note indicating that ~~LCO~~ 3.0.3 does not apply. If moving recently irradiated ~~fuel~~ assemblies while in MODE 1, 2, or 3, ~~the~~ fuel movement is independent of reactor ~~operations~~. Therefore, inability to suspend movement ~~of~~ recently irradiated fuel assemblies is not ~~sufficient~~ reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, ~~movement of recently irradiated fuel,~~ and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the Required Actions of the LCOs for these associated required features. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, ~~movement of recently irradiated fuel assemblies in the primary containment and fuel handling building~~ and operations with a potential for draining of the reactor vessel).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal – shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level – Irradiated Fuel

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft 9 inches above the top of the RPV flange. During refueling, this maintains a sufficient water level in the upper containment pool. Sufficient water is necessary to retain ~~iodine~~ halogen (e.g., iodine) fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient ~~iodine~~ halogen activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR ~~100~~ 50.67 limits, as provided by the guidance of Reference ~~3~~ 5.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. ~~2~~ 5). A minimum water level of 23 ft allows a decontamination factor of ~~100~~ 200 to be used in the accident analysis for ~~iodine~~ halogens. This relates to the assumption that ~~99%~~ 99.5% of the total ~~iodine~~ halogens released from the pellet to cladding gap of all the dropped fuel assembly rods ~~is~~ are retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10%~~ 8% of the total fuel rod ~~iodine~~ I-131 inventory and 5% of the other halogens (Ref. ~~1~~ 5).

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft over irradiated assemblies in the RPV and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the ~~iodine~~ release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4).

While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped

(continued)

BASES

INFORMATION ONLY

Enclosure 6
PY-CEI/NRR-2609L
Page 65 of 69

APPLICABLE SAFETY ANALYSES (continued)

assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases. Based on this judgment, and the physical dimensions which preclude normal operation with water level 23 feet above the flange, a slight reduction in this water level is acceptable.

RPV water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

A minimum water level of 22 ft 9 inches above the top of the RPV flange is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~2~~ 5.

APPLICABILITY

LCO 3.9.6 is applicable during movement of irradiated fuel assemblies within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. Requirements for handling of new fuel assemblies or control rods (where water depth to the RPV flange is not of concern) are covered by LCO 3.9.7, "RPV Water Level — New Fuel or Control Rods." Requirements for fuel handling accidents in the spent fuel storage pools and upper fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level."

ACTIONS

A.1

If the water level is < 22 ft 9 inches above the top of the RPV flange, all operations involving movement of irradiated fuel assemblies within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft 9 inches above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis

(continued)

INFORMATION ONLY

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. ~~Regulatory Guide 1.25, March 1972.~~ Deleted
2. USAR, Section 15.7.6.
3. ~~NUREG-0800, Section 15.7.4.~~ Deleted
4. ~~10 CFR 100.11.~~ 10 CFR 50.67
5. Regulatory Guide 1.183, July 2000

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level – New Fuel or Control Rods

INFORMATION ONLY

BASES

BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain ~~iodine~~ halogen (e.g., iodine) fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient ~~iodine~~ halogen activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR ~~100~~ 50.67 limits, as provided by the guidance of Reference ~~2~~ 5.

APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. ~~1~~ 5). A minimum water level of 23 ft allows a decontamination factor of ~~100~~ 200 to be used in the accident analysis for ~~iodine~~ halogens. This relates to the assumption that ~~99%~~ 99.5% of the total ~~iodine~~ halogens released from the pellet to cladding gap of all the dropped fuel assembly rods ~~is~~ are retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10%~~ 8% of the total fuel rod ~~iodine~~ I-131 inventory and 5% of the other halogens (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft over irradiated assemblies in the RPV and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4).

The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

(continued)

INFORMATION ONLY

Enclosure 6
PY-CEI/NRR-2609L
Page 68 of 69

APPLICABLE SAFETY ANALYSES (continued)	RPV water level satisfies Criterion 2 of the NRC Policy Statement.
LCO	A minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~2~~ 5
APPLICABILITY	LCO 3.9.7 is applicable when moving new fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) over irradiated fuel assemblies seated within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel storage pools and upper fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level." Requirements for handling irradiated fuel over the RPV are covered by LCO 3.9.6, "Reactor Pressure Vessel (RPV) Water Level – Irradiated Fuel."
ACTIONS	<u>A.1</u> If the water level is < 23 ft above the top of irradiated fuel assemblies seated within the RPV, all operations involving movement of new fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.7.1</u> Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is

(continued)

BASES

INFORMATION ONLY

Enclosure 6
PY-CEI/NRR-2609L
Page 69 of 69

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. ~~Regulatory Guide 1.25, March 1972.~~ Deleted
2. USAR, Section 15.7.6.
3. ~~NUREG-0800, Section 15.7.4.~~ Deleted
4. ~~10 CFR 100.11.~~ 10 CFR 50.67
5. Regulatory Guide 1.183, July 2000

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.12 - (Revision 1 - 4/74;RRRC Cat. 4)		
Instrumentation for earthquakes	PNPP design conforms to this guide with the exception of Paragraph C.4.b, Response Spectrum Recorder Frequency Range. The Perry Nuclear Power Plant Response Spectrum Recorders have a frequency range of "2 Hz to 25.4 Hz," rather than the recommended 1 Hz to 30 Hz.	3.7.4
1.13 - (Revision 1 - 12/75;RRRC Cat. 4)		
Spent fuel storage facility design basis	PNPP design conforms to this guide, WITH THE EXCEPTION OF PARAGRAPH C.4. THE INVENTORY OF RADIOACTIVE MATERIALS AVAILABLE FOR LEAKAGE ARE BASED ON THE ASSUMPTIONS GIVEN IN REGULATORY GUIDE 1.183.	~~6.5.1~~, 9.1, 9.4.2
1.14 - (Revision 1 - 8/75)		
Reactor coolant pump flywheel integrity	Not applicable to PNPP design.	-
1.15 - (Revision 1 - 12/72;RRRC Cat. 1)		
Testing of reinforcing bars for Seismic Category I concrete structures	PNPP design conforms to this guide.	3.8.1, 3.8.3, 3.8.4, 3.8.5

INFORMATION ONLY

Enclosure 7
PY-CEI/NRR-2609L
Page 1 of 76

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.24 - (Revision 0 - 3/72;RRRC Cat. 1)		
Assumptions used for evaluating the potential radiological consequences of a pressurized water reactor gas storage tank failure	Not applicable to PNPP design.	-
1.25 - (Revision 0 - 3/72;RRRC Cat. 1)	Not applicable to PNPP. See R.G. 1.183	
Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors	~~PNPP design conforms to this guide with the following exceptions: a. (Regulatory Position C.1.j) filter efficiencies of 95% are used in accordance with Regulatory Guide 1.52; b. (Regulatory Position C.3.a/c) dose conversion factors and average gamma energies are taken from NRC TACT III and/or TACT 5 computer code in lieu of Table 1 and Reference 12.~~	~~6.5.1, 9.1.2, 9.4.2, 15.7.4~~
1.26 - (Revision 3 - 2/76;RRRC Cat. 1)		
Quality group classifications and standards for water-, steam- and radioactive-waste-containing components of nuclear power plants	PNPP design complies with this guide.	3.2.1, Table 3.2-1, 6.2.4, 6.5, 6.7, 9.4, 9.5, 10.3.3, 17.2

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.52 - (Revision 2 - 3/78;RRRC Cat. 2)		
Design, testing and maintenance criteria for postaccident engineered-safety-feature atmosphere cleanup system air filtration and absorption units of light-water-cooled nuclear power plants	PNPP design and testing conform to this guide as presented in Tables 6.5-1 ~~through~~ AND 6.5-3. THE FUEL HANDLING BUILDING FILTER PLENUM HAS BEEN EVALUATED FOR COMPLIANCE TO R.G. 1.140	6.4, 6.5.1, ~~9.1,~~ 9.4, 12.3, 15.7, Tech. Specs.
1.53 - (Revision 0 - 6/73;RRRC Cat. 1)		
Application of single failure criterion to nuclear power plant protection systems	Single failure criteria is applied to protection systems in accordance with Regulatory Guide 1.53.	6.5.3, 7.2.2, 7.3.2, 7.4.2, 7.6.2, 8.1, 9.4
1.54 - (Revision 0 - 6/73;RRRC Cat. 1)		
Quality Assurance requirements for protective coatings applied to water-cooled nuclear power plants	See Chapter 17.2	6.1.1, 6.1.2, 17.2
1.55 - (Revision 0 - 6/73;RRRC Cat. 1)		
Concrete placement in Category I structures	See Chapter 17.2	3.8, 17.2

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.163 (Continued)	The containment isolation check valves in the Feedwater penetrations are tested per the Inservice Testing Program.	
<u>8.1 - (Revision 0 - 2/73)</u> Radiation symbol	PNPP conforms to this guide.	
<u>1.183 - (REVISION 0 - 7/2000)</u> ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS	PNPP CONFORMS TO THIS GUIDE FOR THE FUEL HANDLING ACCIDENT WITH THE FOLLOWING EXCEPTIONS: • APPENDIX B, SECTION 2; WATER DEPTH ABOVE REACTOR FLANGE INSIDE CONTAINMENT IS LESS THAN 23' • APPENDIX B, SECTION 4. THE RADIOACTIVITY THAT ESCAPES FROM THE POOL TO THE BUILDING IS ASSUMED TO BE RELEASED TO THE ENVIRONMENT INSTANTANEOUSLY	9.1.2 9.4.2 15.7.4 15.7.6 TECH SPECS

INFORMATION ONLY

Enclosure 7
PY-CEI/NRR-2609L
Page 4 of 76

3.11.4.4 Diesel Generator Building Ventilation System

The diesel generator building ventilation system is designed to maintain the area at or below 120°F when the diesel generators are operating and outdoor temperature is 95°F. Relative humidity corresponding to this temperature will be less than 50 percent since the building load will consist almost entirely of sensible heat.

Each diesel generator is provided with an independent ventilation system consisting of outdoor air intake louvers and two redundant ventilating fans. These ventilation systems are designed to satisfy Safety Class 3, Seismic Category I requirements and are supplied with onsite emergency power from the associated standby diesel generators should loss of offsite power occur (see Section 9.4.5). No single failure can result in loss of cooling in more than one diesel generator room.

3.11.4.5 Fuel Handling Area Ventilation System

The fuel handling area ventilation system is designed to maintain the area at or below 120°F with an outdoor temperature of 95°F. ~~and to mitigate the consequences of a fuel handling accident~~. The redundant ventilation systems are designed to satisfy Safety Class 3, Seismic Category I requirements. They may be supplied by operator action with onsite emergency power from the standby diesel generators should loss of offsite power occur (see Section 9.4.2). No single failure can result in loss of ventilation to the fuel handling area.

3.11.4.6 Emergency Closed Cooling Pump Area HVAC

The emergency closed cooling pump area HVAC is designed to maintain the area at or below 104°F with both Unit 1 and 2 ECC pumps running, piping heat loss, and instrument air compressors, service air compressors, control complex chillers, and chilled water pumps running. The redundant HVAC units are designed to satisfy Safety Class 3, Seismic

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 ENGINEERED SAFETY FEATURES (ESF) FILTER SYSTEMS

The control room emergency recirculation system (CRERS), ~~the exhaust subsystem of Fuel Handling Area Ventilation System (FHAVS) known as the Fuel Handling Area Exhaust Subsystem (FHAES)~~ and the annulus exhaust gas treatment system (AEGTS) are the ESF filter systems that reduce the concentration of airborne radioactive contaminants following a design basis accident (DBA).

6.5.1.1 Design Bases

Design bases for the charcoal adsorber plenums of the CRERS ~~FHAES~~ and the AEGTS are as follows:

a. Design Criteria

The CRERS ~~FHAES~~ and AEGTS are safety-related. System design conforms with the requirements of General Design Criteria (GDC) 1, 2, 3, 4, 19, AND 60, ~~and 61~~ of Appendix A to 10 CFR 50. To satisfy the requirements of these GDCs, the guidance presented in Regulatory Guides 1.3, ~~1.13,~~ 1.26, 1.29, 1.47, and 1.52 has been considered in the design of these systems.

b. Need for Filtration

The remote possibility of airborne radioactive contaminants entering the control room following a LOCA and the requirements of GDC 19 establish the need for the CRERS for filtration of control room air. GDC 19 requires, in part, that adequate radiation protection be provided to permit access to, and occupancy of, the control room under accident conditions for the duration of the accident without radiation exposure to personnel in excess of 5 rem, whole body (5 rem TEDE for the design basis LOCA).

~~The remote possibility of release of airborne radioactive contaminants due to a fuel handling accident, the requirements of GDC 61, and the recommendations of Regulatory Guides 1.13 and 1.25 establish the need for the FHACES to accomplish fuel pool area air filtration. GDC 61 requires, in part, that fuel storage and handling, and radioactive waste and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions and that appropriate filtering systems be provided.~~

The AEGTS is provided to reduce the radiological consequences of fission product releases in the containment from a LOCA~~, or fuel handling accident involving recently irradiated fuel~~ by collecting and filtering the leakage from containment. Also, the AEGTS is designed to maintain a negative pressure in the annulus relative to the outside which minimizes ground level release of airborne radioactivity due to containment exfiltration during normal and postaccident conditions.

c. Component System Sizing

Two 100 percent capacity filter units are provided for the CRERS. Air flow rate for the CRERS is 30,000 cfm per plenum. Based on this assumed air flow rate and the assumed charcoal adsorber efficiencies and factors discussed in Section 15.6, the overall dose to the operators following an accident has been shown to satisfy the requirements of GDC 19, or the 5 rem TEDE dose limit used for the design basis LOCA dose calculations.

INFORMATION ONLY

~~Three 50 percent capacity filter units are provided for the FHAES. The FHAES provides exhaust flow from the fuel handling area, the fuel pool cooling equipment rooms, the control rod drive maintenance area, and the control rod drive pump areas. Flow is 30,000 cfm. Of this quantity, 15,300 cfm is exhausted directly from the fuel pool area. This air flow rate is based on flow patterns that should entrain contaminants escaping from the fuel pool area.~~

Two 100 percent capacity AEGTS filter units are provided for each reactor unit. Air flow rate for the AEGTS is 2,000 cfm per plenum. Based on this flow rate, the negative pressure in the annulus is maintained at -0.25 inches of water gauge minimum continuously.

Components of these filter systems have been sized to handle system air flow based on the recommendations of Regulatory Guide 1.52, ERDA 76-21 and general engineering practice.

d. Fission Product Removal Capability

The fission product removal capability of the activated charcoal adsorber material used in the CRERS, ~~FHAES~~ and AEGTS is based on the recommendations of Regulatory Guide 1.52.

The decontamination efficiency of the AEGTS charcoal adsorber is 99 percent for both elemental iodine and organic species of iodine. For the revised accident source term LOCA analysis, no credit was taken for the removal of elemental and organic iodines by the charcoal filters in the AEGTS. The AEGTS charcoal adsorber bed is 4 inches deep with annulus exhaust air maintained at less than 70 percent relative humidity.

The decontamination efficienc~~ies~~y used for the CRERS ~~and FHAES~~ charcoal adsorber~~s~~ is ~~are~~ 99 percent for elemental iodine and 95 percent for organic species of iodine. For the revised accident source term LOCA analysis and a Fuel Handling Accident sensitivity case, an elemental and organic removal efficiency of 50% was assumed for the charcoal filters in the CRERS. The CRERS ~~and FHAES~~ charcoal adsorber bed~~s~~ is ~~are~~ 2 inches deep. Exhaust air for ~~both~~ the CRERS plenum~~s~~ is maintained at less than 70 percent relative humidity.

The HEPA filter efficiency used for all the plenums is 99.97 percent on particles 0.3 microns and larger.

Additional bases for the design of the CRERS ~~FHAES~~ and AEGTS are presented in Sections 6.4 ~~9.4.2~~ and 6.5.3, respectively.

6.5.1.2 System Design

The design features of the CRERS ~~FHAES~~ and AEGTS are compared to the recommendations of Regulatory Guide 1.52 in Tables 6.5-1 ~~6.5-2~~ and 6.5-3, respectively.

Design of the activated charcoal adsorber plenums used in the CRERS ~~FHAES~~ and AEGTS follows the guidelines of Regulatory Guide 1.52 and ERDA 76-21.

Each charcoal adsorber plenum contains the following:

a. Demisters to remove large particles and water droplets (about 1 micron diameter).

b. Roughing filters to remove large particles (about 1 micron).

c. HEPA filters to remove small particles (0.3 to 1 micron), including fission product aerosols (particulates).

Roughing and HEPA filters are replaced when pressure drop across a filter exceeds the technical specification value. Pressure drop is measured by permanently installed differential pressure indicators. Charcoal adsorber material is changed when laboratory test results from representative samples show that the adsorber fails to satisfy the requirements of the Ventilation Filter Test Program.

Essential services, such as power and electrical control cables associated with ESF filter systems, are protected as described in Section 8.3.1.4.

The charcoal adsorber portion of each filter train is provided with a high temperature detection and water spray system to allow flooding of the charcoal bed in the unlikely event of high temperature in the charcoal (to preclude the possibility of iodine desorption).

6.5.1.3 Design Evaluation

SIMILAR CHANGES MADE THROUGHOUT SECTION 6.5

Design and safety evaluations of the CRERS/ ~~FHAES~~ and AEGTS are presented in Sections 6.4/ ~~9.4.2~~ and 6.5.3, respectively.

The charcoal adsorber plenums are not exposed to conditions that can impair plenum efficiency. The exhaust air flowing through the charcoal plenums is maintained at 70 percent relative humidity by operation of the electric heating coils during abnormal conditions.

The ~~FHAES and~~ AEGTS is ~~are~~ normally operated continuously during plant operation.

The CRERS is operated for at least 10 hours each month as recommended by Regulatory Guide 1.52; during this operation of CRERS, the exhaust air is free of radioactive contaminants. Air exhausted or

TABLE 6.5-2

COMPARISON OF FUEL HANDLING AREA EXHAUST SUBSYSTEM WITH REGULATORY GUIDE 1.52 POSITIONS

Regulatory Position	System Design Feature
1.a	The design conforms with this position.
1.b	The design conforms with this position.
1.c	The design conforms with this position.
1.d	The design conforms with this position.
1.e	The design conforms with this position.
2.a	The design conforms with this position.
2.b	The design conforms with this position.
2.c	The design conforms with this position.
2.d	The filter units are not exposed to pressure surges from the postulated DBA.
2.e	The design conforms with this position.
2.f	The design conforms with this position.
2.g	Pressure drops and flow rates are not monitored in the control room.
2.h	The design conforms with this position.
2.i	The design conforms with this position.
2.j	The design conforms with the intent of this position. See Section 12.1 for a discussion of conformance with Regulatory Guide 8.8.
2.k	The design conforms with this position.
2.l	The design conforms with this position. Duct and housing leak tests will be performed in accordance with Section 6 of ANSI N510-1980 instead of ANSI N510-1975

All of this Table is removed

TABLE 6.5-5

FUEL HANDLING AREA EXHAUST SUBSYSTEM

MATERIALS LIST (DESIGN DATA)

Filter Unit Housing	
Number of filter units	3 for Units 1 & 2
Manufacturer	CVI-Pennwalt
Filter holding frame	Stainless steel, ASTM A479, Type 304
Demisters	
Number per filter unit	12
Manufacturer & model no.	ACS, #101-55
General standards	MSAR-71-45
Frame material	Stainless steel, ASTM A479, Type 304
Media	Stainless steel and fiberglass woven mesh
Roughing Filters	
Number, per filter unit	12
Manufacturer	Flanders (or equal)
Model number	00A-0-02-03NL
General standards	UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material
Adhesives	Fire retardant polyurethane and rubber based adhesives

TABLE 6.5-5 (Continued)

Gaskets	Sponge Neoprene (SCE-43)
Weight, per filter, lbs	
Steel, approx.	25
Glass and miscellaneous material, approx.	10
Total	35
<u>HEPA Filters Upstream and Downstream</u>	
Number, per filter unit	24
Manufacturer	Flanders (or equal)
Model number	007-0-02-03NU
General standards	MIL-F-51068D; UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material (MIL-F-51079)
Adhesives	Fire retardant solid urethane
Gaskets	Sponge Neoprene (SCE-43) (ASTM D1056)
Weight, per filter, lb	
Steel, approx.	25
Glass and miscellaneous material, approx.	15
Total	40

INFORMATION ONLY

TABLE 6.5-5 (Continued)

Activated Charcoal Adsorber	
Manufacturer	CVI-Pennwalt
Type of media	New, activated coconut shell charcoal
Impregnant	Potassium iodide (KI) and elemental iodine type
Weight of carbon, lb	3,040
Adsorber enclosure	Stainless steel, ASTM A240, Type 304
Electric Heating Coil	
Manufacturer	CVI-Pennwalt
Frame material	Stainless steel
Heating element	Inconel steel sheathed elements

TABLE 7.1-1

DESIGN AND SUPPLY RESPONSIBILITY OF SAFETY-RELATED SYSTEMS

	GE Design	GE Supply	Others
Reactor Protection Trip System			
Reactor Protection Trip System (RP)	X	X	
Engineered Safety Features Systems			
Emergency Core Cooling Systems (ECCS)	X	X	
High Pressure Core Spray (HPCS)			
Automatic Depressurization System (ADS)			
Low Pressure Core Spray System (LPCS)			
RHR Low Pressure Coolant Injection (LPCI)			
Containment and Reactor Vessel Isolation Control System (CRVICS)	X	X	X
Process Radiation Monitoring System (PRM) (Portion used for CRVICS)	X	X	X
Emergency Water Systems			
Emergency Closed Cooling Water (ECCW)			X
Emergency Service Water (ESW)			X
Control Complex Heating Ventilation and Air Condition System			X
Combustible Gas Control System			X
Annulus Exhaust Gas Treatment System (AEGTS)		X	
ESF Building and Area HVAC and Purification System			X
Containment Vacuum Relief System			X
Suppression Pool Makeup System			X
RHRS Containment Spray Cooling Mode	X	X	
RHRS Suppression Pool Cooling Mode	X	X	
Standby Power Systems	X	X	X
Pump Room Cooling Systems			X
~~Fuel Handling Ventilation System~~			~~X~~
Systems Required for Safe Shutdown			
Standby Liquid Control System (SLCS)	X	X	
RHR Reactor Shutdown Cooling Mode	X	X	
Remote Shutdown System (RSS)	X	X	X
Reactor Core Isolation Cooling System (RCIC)	X	X	

TABLE 7.1-1 (Continued)

	GE Design	GE Supply	Others
Safety-Related Display Instrumentation	X	X	X
All Other Safety-Related Systems			
Process Radiation Monitoring System			X
Neutron Monitoring System	X	X	
Intermediate Range Monitor (IRM)			
Average Power Range Monitor (APRM)			
Local Power Range Monitor (LPRM)			
Leak Detection	X	X	X
Rod Pattern Control System (RPCS)	X	X	
Recirculation Pump Trip (RPT)	X	X	
Fuel Pool Cooling System (FPCS)			X
Offgas Building Exhaust			X
Containment Atmosphere Monitoring System			X
High Pressure - Low Pressure Systems Interlocks	X	X	
Redundant Reactivity Control System	X	X	
Hydrogen Control System			X
FUEL HANDLING VENTILATION SYSTEM			X

7.3 ENGINEERED SAFETY FEATURE SYSTEMS

7.3.1 DESCRIPTION

Section 7.3 describes the instrumentation and controls of the following plant Engineered Safety Features (ESF) systems:

a. Emergency Core Cooling Systems (ECCS)

b. Containment and Reactor Vessel Isolation Control Systems (CRVICS)

c. (Deleted)

d. RHRS-Containment Spray Cooling Mode (RHRS-CSCM)

e. RHRS-Suppression Pool Cooling Mode (RHRS-SPCM)

f. Emergency Water Systems (EWS)(1)

g. Control Complex HVAC System(1)

h. ESF Building and Area HVAC System(1)

i. Annulus Exhaust Gas Treatment System (AEGTS)

j. Pump Room Cooling System(1)

k. Containment Combustible Gas Control System

l. Suppression Pool Makeup System

m. Containment Vacuum Relief

No changes. Included for completeness

n. Standby Power Support Systems[1]

~~o. Fuel Handling Area Exhaust Subsystem[2]~~

NOTE:

1. The following systems are considered to be ESF support systems not ESF systems in accordance with the guidance provided in NUREG-0800, Section 7.3. These systems will continue to be treated as safety-related for design, construction, maintenance, testing, and other operational purposes. Independent actuation of any one of these systems will not be reported per 10 CFR 50.73(a)(2)(iv).

   a. Emergency Closed Cooling Water (ECC) (P42)

   b. Control Complex Chilled Water (CCCW) (P47)

   c. ESF Building and Area HVAC Systems (M23)(M24)(M43)

   d. Pump Room Cooling Systems (M28)(M32)(M39)

   e. Standby Power Support Systems (R44)(R45)(R46)(R47)(R48)

~~2. Only the exhaust subsystem of the fuel handling area ventilation system is ESF.~~

The sources which supply power to the engineered safety feature systems originate from onsite ac and/or dc safety-related busses or, as in the case of the CRVICS failsafe logic, from the nonsafety-related RPS MG sets. Refer to Chapter 8 for a complete discussion of the ESF systems power sources.

engine cooling water system, these temperatures may be displayed on the digital temperature indicator on the local control panel.

Another set of thermocouples in the lubricating oil piping feed oil temperature in and out of the engine signals to a slow speed temperature recorder in the local control panel. This recorder operates continuously and provides a continuous record of important engine temperature for performance monitoring, trending and engine diagnostics.

4. Diesel Generator Cooling Water System

The diesel engine cooling water system is designed to remove the heat loads of the engine air intercooler, oil cooler and water jacket. Additional information on this system is provided in Section 9.5.5 for the standby diesel generators and Section 9.5.9.2 for the HPCS diesel generators.

~~7.3.1.1.16 Fuel Handling Area Exhaust Subsystem~~

~~The Fuel Handling Area Exhaust Subsystem (FHAES) is a subsystem of the Fuel Handling Area Ventilation System (FHAVS). The FHAES is an ESF System.~~

~~a. FHAES Function~~

~~The purpose of the exhaust subsystem is to exhaust air from potentially contaminated areas. The air is filtered and passed through a charcoal filter train prior to discharge to atmosphere via the unit vent.~~

The remainder of the discussion of the FHAES is also removed (through page 7.3-62.

(b) RCIC Steam Admission valve Open. (RCIC Pump Room only)

16. Control Complex HVAC

(a) Reactor Vessel Low Water Level (Trip Level 1)

(b) Drywell High Pressure

(c) High Radiation

(d) Loss of Offsite Power

~~17. Fuel Handling Area Ventilation System~~

~~(a) Charcoal Filter Inlet High Radiation~~

The plant conditions which require protective action involving the ESF systems are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

Where applicable in Technical Specifications, the minimum number of sensors is specified to monitor safety-related variables. There are no sensors in the ESF systems which have a spatial dependence.

c. Prudent Operational Limits

Operational limits for each safety-related variable trip setting are selected with sufficient margin so that a spurious ESF system initiation is avoided. It is then verified by analysis that the

A radiologicAL analysis, as a result of a fuel drop from 10 feet above the racks, IS BOUND BY THE ANALYSIS FOR A FUEL DROP INSIDE CONTAINMENT ~~has been performed~~. It identifies the causes of the accident, assumptions and starting conditions. The fission product release from the fuel, the airborne activity ~~in the fuel handling area of the intermediate building and~~ to the environs, is calculated and the corresponding radiological effects offsite are evaluated using the methods, assumptions and conditions in Regulatory Guide~~s~~ 1.183 ~~1.25 and 1.52~~. The discussion of the consequences of this accident are presented in Section 15.7.4.

9.1.2.3.3 Spent Fuel Rack Design - GE Racks

Spent fuel rack design features are as follows (refer also to Figure 9.1-2):

a. Each containment spent fuel pool contains 19 sets of racks which may contain up to 190 fuel assemblies. A maximum of 380 fuel assemblies may be stored in the two spent fuel pools.

b. The storage racks provide an individual storage compartment for each fuel assembly and are secured to the pool wall through associated hardware. The fuel assemblies are stored in a vertical position with the lower tie plate engaged on a captive slot in the lower fuel rack support casting. Additional restraints are provided to restrict lateral movement.

c. The weight of the fuel assembly is held by the lower rack support casting.

d. The spent fuel storage racks are made from aluminum. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the

3. Three-way solenoid valves to automatically close (open) the control air supply port to the pneumatic driven dampers when the corresponding air handling unit is stopped (started).

4. Temperature transmitters, temperature indicators and pneumatic temperature controllers in the computer rooms to modulate the three-way valve controlling chilled water flow through the air handling unit cooling coils.

5. Humidity controllers in the computer rooms to modulate the humidifier control valve controlling the amount of steam discharge to the supply air stream.

6. High and low limit moisture switches that provide a signal to alarm the control room of high or low relative humidity condition in the computer rooms. High limit moisture switches also automatically close steam supply motor operated valves at the same high setpoint and automatically open the valves when the relative humidity is low.

7. Temperature element in each computer room to alarm in the control room if the room temperature rises above the high temperature setpoint.

No changes to this page. Included for information

9.4.2 FUEL HANDLING AREA VENTILATION SYSTEM

The fuel handling area ventilation system (FHAVS) is comprised of the fuel handling area supply subsystem (FHASS) and the fuel handling area exhaust subsystem (FHAES). These systems provide ventilation for the general fuel handling area, fuel pool area, control rod drive pump areas, and the fuel pool cooling equipment room.

9.4.2.1 Design Bases

Design bases for the FHASS and the FHAES are as follows:

a. The FHASS and FHAES are classified as Safety Class 3, Seismic Category I. System design complies with the requirements of General Design Criteria (GDC) 1, 2, 4, 5, 60, and 61 of 10 CFR 50, Appendix A. The guidance provided by Regulatory Guides 1.3, 1.13, ~~1.25~~, 1.26, 1.29, 1.47, ~~and~~ ~~1.52~~ 1.140, AND 1.183 has been considered in the system design.

b. The FHASS and FHAES are not required for safe shutdown of the plant in the event of a LOCA. ~~However,~~ The FHAES is NOT CREDITED ~~required~~ to operate to mitigate the consequences of a fuel handling accident involving ~~recently~~ irradiated fuel bundles. ~~(i.e., fuel that has occupied part of a critical reactor core within the previous seven days)~~. ~~Therefore,~~ Redundant components are provided to satisfy the single failure criterion. ~~The FHAES is an ESF system.~~

c. The FHASS and FHAES are initially started and subsequently operated remote-manually from the control room.

d. The FHASS and FHAES are designed to maintain the temperature of the fuel handling areas, and any other areas they serve, between the temperatures given in Table 3.11-4. This temperature range is suitable for operating personnel and equipment.

e. The FHASS and FHAES are designed so that air flow is directed from areas of low probable airborne contamination to areas of high probable airborne contamination.

f. ALTHOUGH NOT CREDITED IN THE FUEL HANDLING ACCIDENT ANALYSIS, the FHAES passes exhaust air from the fuel handling area through charcoal filter trains to ensure that release of radioactivity to the environment is kept below permissible discharge limits.

g. The FHASS outside air intake is provided with a structural missile barrier to prevent external missiles from entering the fuel handling area. The air intake duct is provided with two barometric pressure relief dampers. The system discharges the exhaust air to the atmosphere through a concrete vent which provides a structural barrier against external missiles.

h. The major components of the FHAVS are physically separated and located so that they are not affected by internally generated missiles, pipe whip and jet impingement forces associated with breaks in high and moderate energy piping.

i. DESIGN FEATURES OF THE FHAES IS ALSO CONTAINED IN ~~SECTION 6.5.1.2~~ TABLE 12.3-3.

9.4.2.2 System Description

Ventilation of the fuel handling area and other associated areas is accomplished by the FHAVS. This system is shown schematically on Figure 9.4-4.

The FHAVS is designed to provide heating and ventilation for the various operating areas of the fuel handling area and ventilating equipment areas, and to provide effective protection for personnel against airborne radioactive contaminants.

The FHASS continuously draws outside air through roughing filters and heating coils. One of the two 100 percent capacity supply fans (M40-C001 A, B) normally operates to draw air through the supply plenum and discharge it to the supply ductwork for distribution. The areas provided with supply air are the control rod drive pump areas, the FHAVS equipment area, the railway and overhead crane area, and the periphery of the fuel pool area on all sides. The general air flow pattern in the fuel handling area is from areas of low probable airborne contamination to areas of high probable airborne contamination.

The ventilation pattern in the fuel pool areas is from the supply around the periphery of the pools toward the exhaust located directly above the pools.

The FHAES continuously draws air from the CRD pump areas, the control rod drive maintenance area, the area above the fuel pools, and from the fuel pool cooling, cleaning and postaccident sampling system (PASS) equipment rooms located in the intermediate building. Two of the three 50 percent charcoal exhaust units are operating normally to draw air through the exhaust ductwork and discharge it to the atmosphere through the unit vent.

In the event that the radiation monitors located upstream of the charcoal exhaust units senses high radiation, the high radiation signal alarms in the control room and automatically trips the supply fan. The exhaust system remains operational to continue exhausting contaminated air from the fuel handling area through charcoal filters, thus precluding any uncontrolled release of radioactivity to the outside. Two barometric pressure relief dampers (F575 and F576) in the supply duct would relieve any excessive negative building pressure. THIS FILTRATION IS NOT CREDITED IN THE FUEL HANDLING ACCIDENT ANALYSIS

During normal plant operation and plant shutdown, power will be provided by the preferred ac source. In case of a LOCA, this system is not required to operate to safely shut down the plant. However, during loss of offsite power (without LOCA), this system is automatically connected to the diesel generator and may be started manually at the operator's option.

To comply with the single failure criterion, the power for exhaust fan M40-C002C and filter train M40-D001C is provided from Division 1 or 2, preferred ac sources. The division transfer at the motor control center is done by a manual key interlock system. An effective means to maintain cable and wiring separation between Division 1 and 2 is radiological considerations during normal operation and a fuel drop accident are discussed in Section 12.2 and Chapter 15.0, respectively. ~~Comparison of the fuel handling building exhaust air filtration system with the positions in Regulatory Guide 1.52 is given in Section 6.5.1.~~

9.4.2.4 Inspection and Testing Requirements

The components of the fuel handling building ventilation supply and exhaust subsystems are accessible during normal plant operation, shutdown and refueling operations. The ability to isolate an idle redundant component enables inspection, maintenance and testing to be performed while the system is in normal operation. When maintenance and testing are required on the common supply plenum, the supply system will be shut down.

Periodic tests will be performed on the fuel handling building exhaust filter system. These tests will include measurement of differential pressure across the filter units and determination of filter efficiency to demonstrate that aging, weathering or poisoning of the filters has not significantly degraded the adsorptive material in the charcoal and HEPA filters. ~~Section 6.5.1 gives additional testing requirements for the charcoal filter trains.~~

During testing and inspection, provision will be made to verify the function and performance of the fans, dampers, valves, controls, and other safety devices to ensure that these operational components perform their function reliably and accurately during normal operation, and under conditions of operating interruptions.

9.4.2.5 Instruments, Controls, Alarms, and Protective Devices

Operation of the fuel handling area ventilation supply and exhaust subsystems is manually initiated from the control room. During operation, one of the two supply fans and two of the three exhaust fans

An audible alarm and an alarm acknowledge pushbutton is provided at the local panel. The acknowledge pushbutton silences the local audible alarm and clears the control room system trouble alarm.

5. Radiation monitor in the heater bay vent stack to alarm in the control room on detection of high radioactivity level in the exhaust air.

9.4.4.5.3 Offgas Building Exhaust System

a. Operation of this system is initiated manually from the control room. During normal operation, one of the two fans operate continuously. Details of the instrumentation and controls for this system are discussed in Section 7.6.1.

b. The operation of the offgas holdup pipe room fan is initiated manually from wall panel 1H51-P5236 next to the offgas holdup pipe room entrance. During normal operation of the offgas system this fan would operate continuously.

No changes to this page. Included for information

9.4.5 ENGINEERED SAFETY FEATURES VENTILATION SYSTEM

The engineered safety features (ESF) ventilation systems discussed in this section are the emergency service water pump house ventilation system, emergency closed cooling pump area cooling system, ECCS pump room cooling systems, and the diesel generator building ventilation system. Additional ESF ventilation systems are discussed in the sections noted:

a. Annulus exhaust gas treatment system Section 6.5.3

b. Control room HVAC and control room emergency recirculation system — Section 6.4

c. MCC, switchgear and miscellaneous electric equipment areas HVAC/battery room exhaust system — Section 9.4.1

d. ~~Fuel handling area exhaust subsystem~~ — ~~Sections 6.5.1 and 9.4.2~~

e. Control complex chilled water system — Section 9.4.9

9.4.5.1 Design Bases

9.4.5.1.1 Emergency Service Water Pumphouse Ventilation System

Design bases for the Emergency Service Water Pumphouse Ventilation System (ESWVS) are as follows:

a. The ESWVS is classified as Safety Class 3, Seismic Category I. The design of this system complies with the requirements of General Design Criteria (GDC) 1, 2, 3, 4, and 5 of 10 CFR 50 Appendix A, and 10 CFR 50 Appendix B. The requirements of Regulatory Guides 1.26, 1.29, 1.47, 1.53, National Fire Protection Association (NFPA) 90A, and Branch Technical Position APCSB 9.5-1 have also been considered in the system designs and equipment procurement.

b. The ESWVS is:

    1. Required to operate to safely shut down the plant during normal conditions, and emergency or LOCA conditions.

radiation alarm can be obtained in the control room following the leakage of less than 10 gallons. A plate out factor of 2 is taken into consideration for this evaluation.

With a reactor coolant pressure of 1,000 psi, and a flow restrictor of 1/4 inch, the initial flow rate through the break, assuming 100 percent flashing will be in excess of 10 gpm. Thus a monitor response to the potential hazard of less than 10 minutes is expected.

d. Fuel Handling Building Exhaust:

The drop of a channeled spent fuel bundle has been identified in Section 15.7.4 as a hazard for personnel in this building.

Assuming that ~~7.3~~ 7.49 x 10^2 Ci of Kr^{85} (Table 15.7-~~18~~ 34) are released and mixed instantaneously into the whole volume of the fuel handling building, the resulting concentration is expected to be 1.7 x 10^{-2} μCi/cc. This activity level is well within the range capability of the monitor.

The response time of the monitor is inversely proportional to the activity level and is expected to be negligible at the high anticipated levels which may be reached during this calculable fuel handling accident.

The particulate and iodine filters are removable for laboratory analysis to verify and identify activity levels and to provide a backup to the continual monitoring of the areas of surveillance.

d. Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

15.0.3.1.3 Unacceptable Results for Limiting Faults [Design Basis (Postulated) Accidents]

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

a. Radioactive material release which results in dose consequences that exceed the guideline values of 10 CFR 100 (for the design basis RAST - LOCA analysis, the offsite dose limit is 25 rem TEDE. FOR THE FUEL HANDLING ACCIDENT, THE OFFSITE DOSE LIMIT IS 6.3 REM TEDE).

b. Failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited.

c. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required.

e. Radiation exposure to plant operations personnel in the main control room in excess of 5 rem whole body, 30 rem inhalation and 75 rem skin [5 rem TEDE for the design basis LOCA (RAST analysis) AND FUEL HANDLING ACCIDENT].

15.0.3.2 <u>Sequence of Events and Systems Operation</u>

Each transient or accident is discussed and evaluated in terms of:

a. A step-by-step sequence of events from initiation to final stabilized condition.

For limiting faults (design basis accidents) two quantitative analyses are considered:

a. The first is based on conservative assumptions for the purposes of worst case bounding of event consequences to determine the adequacy of the plant design to meet 10 CFR 100 guidelines (for the design-basis RAST - LOCA Analysis, the licensing basis limit is 25 rem TEDE). This analysis is referred to as the "design basis analysis."
THE FUEL HANDLING ACCIDENT LICENSING BASIS LIMIT IS 6.3 REM TEDE (OFFSITE) AND 5 REM TEDE (CONTROL ROOM)

b. The second is based on realistic assumptions to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis." The "realistic analysis" is not performed for the LOCA-RAST analysis, OR FUEL HANDLING ACCIDENT.

Results for both are shown to be within NRC guidelines.

Doses resulting from the events in Chapter 15 are determined either manually or by computer code. Doses associated with Offgas System Failure (Section 15.7.1.1) are evaluated using GASPAR II (NUREG/CR-4653).[8] Time dependent releases are evaluated with the TACT computer code.[2][6] Instantaneous or "puff" type releases are evaluated by methods based on those presented in Regulatory Guide 1.3, 1.183 ~~and~~ AND NUREG-1465. The General Electric NEDO-31400 analysis (7) also is utilized in determining doses associated with a Control Rod Drop Accident (Section 15.4.9). Dose conversion factors and breathing rates are presented in Table 15.0-4.

15.0.4 NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) RELATIONSHIP

Appendix 15A is a comprehensive, total plant, system-level, qualitative failure modes and effects analysis, relative to all the Chapter 15 events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in protective actions.

TABLE 15.0-3

SUMMARY OF ACCIDENTS

Section	Title	Failed Fuel Pins: GE Calculated Value	Failed Fuel Pins: NRC Worst Case Assumption
15.3.3	Seizure of One Recirculation Pump	None	
15.3.4	Recirculation Pump Shaft Break	None	
15.4.9	Control Rod Drop Accident	<770	770
15.6.2	Instrument Line Break	None	None
15.6.4	Steam System Pipe Break Outside Containment	None	None
15.6.5	LOCA Within RCPB	None	100%
15.6.6	Feedwater Line Break	None	None
15.7.1.1	Main Condenser Offgas Treatment System Failure	N/A	N/A
15.7.3	Liquid Radwaste Tank Failure	N/A	N/A
15.7.5	Spent Fuel Cask Drop Accident	None	None
15.7.6	Fuel Handling Accident Inside Containment (GE 14 FUEL W/TRIANGULAR MAST)	~~<124~~ 151	~~124~~
15.8	ATWS	SPECIAL EVENT	

TABLE 15.0-4

DOSE CONVERSION FACTORS*

Isotope	Thyroid (rem/Ci)	Whole Body 0.25xMeV/dis
I-131	1.49E+6	8.72E-2
I-132	5.35E+4	5.13E-1
I-133	3.97E+5	1.55E-1
I-134	2.54E+4	5.32E-1
I-135	1.24E+5	4.21E-1
Kr-83m		5.02E-6
Kr-85		3.72E-2
Kr-85m		5.25E-4
Kr-87		1.87E-1
Kr-88		4.64E-1
Kr-89		5.25E-1
Xe-131m		2.92E-3
Xe-133m		8.00E-3
Xe-133		9.33E-3
Xe-135m		9.92E-2
Xe-135		5.72E-2
Xe-137		4.53E-2
Xe-138		2.81E-1

Breathing Rates

Time Period (hr)	Breathing Rate ($m^{(3)}$/sec)
0-8	3.47E-4**
8-24	1.75E-4
24-720	2.32E-4

* The following dose conversion factors (DCF's) are used in the design basis LOCA-RAST analysis, AND FUEL HANDLING ACCIDENT.

- DCF's for inhalation: EPA Federal Guidance Report 11 (Reference 9)

**This breathing rate was used for the duration of the Control Room radiological consequence analyses (LOCA). BREATHING RATE OF 3.5E-4 USED IN THE FUEL HANDLING ACCIDENT CONTROL ROOM ANALYSIS

k. Credit is taken for dilution in the lake to the nearest drinking water intake (0.5 miles ENE of the plant) as presented in Table 5.1-10 of the Perry Nuclear Power Plant Environmental Report (Operating License Stage).

The resulting exposures from liquid releases to the groundwater are presented in Table 15.7-14.

The individual isotopic concentrations and fraction of maximum permissible concentrations (FMPC) for the radionuclides released by a postulated failure of the concentrated waste tank are given in Table 15.7-15a. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the 10 CFR 20 regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised 10 CFR 20 dated October 4, 1993.)
A summary of the total isotopic concentration and total FMPC for each component postulated to fail is given in Table 15.7-16.

As indicated by these results the concentrations are well within the 10 CFR 20 effluent concentration limits for unrestricted areas (10 CFR 20, Appendix B). Likewise, the resultant exposures are a small fraction of acceptable limits for this type of event.

15.7.4 FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT

This accident is not reanalyzed as part of the reload analyses as the ~~initial cycle~~ FUEL HANDLING ACCIDENT INSIDE CONTAINMENT analysis is bounding (SEE 15.7.6). Radiological exposures were recalculated ~~for Cycle 9~~ incorporating GE14 fuel resulting in exposures ~~well below 10 CFR 100 guidelines.~~ WITHIN THE LICENSING BASIS LIMITS OF 6.3 Rem TEDE (OFFSITE) AND 5 REM TEDE (CONTROL ROOM)

INFORMATION ONLY

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto stored fuel bundles.

15.7.4.1.2 Frequency Classification

This event has been categorized as a limiting fault.

No changes to this page. Included for information.

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

The most severe fuel handling accident from a radiological release viewpoint is the drop of a channeled spent fuel bundle onto unchanneled spent fuel in the spent fuel racks in the fuel handling building. The sequence of events which is assumed to occur is as follows:

	Event	Approximate Elapsed Time
a.	Channeled fuel bundle is being handled by a crane over spent fuel pool. Crane motion changes from horizontal to vertical and the fuel grapple releases, dropping the bundle. The channeled bundle strikes unchanneled bundles in the rack.	0
b.	Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
c.	Gases pass from the water to the fuel handling building.	0
d.	The fuel handling building ventilation system high radiation alarm alerts plant personnel.	<1 Min

15.7.4.2.1.1 Identification of Operator Actions

The accident analysis does not assume any operator actions for the mitigation of this event.

No changes to this page. Included for information.

15.7.4.2.2 Systems Operation

~~Normally, operating plant instrumentation and controls are assumed to function although credit is taken only for the operation of the FHAES~~. Operation of ~~other~~ plant, reactor protection or ESF systems is not taken into account.

15.7.4.2.3 The Effect of Single Failures and Operator Errors

The FHAES is designed to single failure criteria and safety requirements. NO CREDIT IS TAKEN FOR THE FHAES.

~~Refer to Sections 7.3 and 9.4, and Appendix 15A for further details~~.

15.7.4.3 Core and System Performance (INITIAL CYCLE)

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences for the initial cycle.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts.

To estimate the expected number of failed fuel rods in each impact, a conservation of energy approach is used. The fuel assembly is expected to impact on the spent fuel racks at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods are expected to absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1 ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure (based on 1 percent uniform plastic deformation of the rods).

The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other pool structures. Because an unchanneled fuel assembly consists of 76 percent fuel, 19 percent cladding and 5 percent other structural material by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

The energy absorption on successive impacts is estimated by considering a plastic impact. Conservation of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is:

$$1 - \frac{M_1}{M_1 + M_2}$$

where M_1 is the impacting mass and M_2 is the struck mass.

15.7.4.3.2 Input Parameters and Initial Conditions (INITIAL CYCLE)

The assumptions used in the analysis of this accident are listed below:

a. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

b. The entire amount of potential energy, referenced to the top of the spent fuel racks, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the rack and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple or the grapple cable breaks.

c. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

15.7.4.3.3 Results (INITIAL CYCLE)

a. Energy Available

Dropping a fuel assembly onto the spent fuel racks from the maximum assumed height of 10 ft (actual height is 8 ft), results in an impact velocity of 25.4 ft/sec.

The kinetic energy acquired by the falling fuel assembly is less than 8,000 ft-lb and is dissipated in one or more impacts.

b. Energy Loss Per Impact

Based on the fuel geometry in the spent fuel rack, two fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is approximately 63 percent.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 22 more fuel assemblies, so that after the second impact only 88 ft-lb (approximately 2 percent of the original kinetic energy), is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rod will fail on a third impact. In calculating the activity release, however, it is conservatively assumed that one rod fails on the third impact.

If the dropped fuel assembly strikes only one fuel assembly on the first impact, the energy absorption by the fuel rack support structure results in approximately the same energy dissipation on the first impact as in the case where two fuel assemblies are struck. The energy relations on the second and third impacts remain approximately the same as in the original case. Thus, the calculated energy dissipation is as follows:

First impact	63 percent
Second impact	35 percent
Third impact	2 percent (no cladding failures)

c. Fuel Rod Failures

No changes.
Provided for information

1. First Impact Failures

The first impact dissipates 0.63 x 8,000 or 5,040 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in rack. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail. Because the 8 tie rods of each struck fuel assembly are more susceptible to bending failure than the other 54 fuel rods, it is assumed that they fail on the first impact. Thus 2 x 8 = 16 tie rods (total in 2 assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the spent fuel racks, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the remaining rods of the 2 struck assemblies, 250 x 54 x 2 or 27,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 \times 5{,}040 \times \frac{19}{19 + 5}}{250} = 8$$

Thus, during the first impact, fuel rod failures are as follows:

Dropped assembly	62 rods (bending)
Struck assemblies	16 tie rods (bending)
Struck assemblies	8 rods (compression)
	86 failed rods

2. Second Impact Failures

No changes
Provided for information

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 22 struck assemblies are the tie rods subjected to bending failure. Thus 2 x 8 = 16 tie rods are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{\frac{0.35}{2} \times 8{,}000 \times \frac{19}{19 + 5}}{250} = 5$$

INFORMATION ONLY

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	16 tie rods (bending)
Struck assemblies	5 rods (compression)
	21 failed rods

3. Total Failures

The total number of failed rods resulting from the accident is as follows:

First impact	86 rods
Second impact	21 rods
Third impact	1 rods
	108 total failed rods (initial cycle)

15.7.4.3.4 RESULTS (CURRENT CYCLE)

~~151 total failed rods are assumed for GE14 fuel~~.
THE TOTAL ~~FAILED~~ NUMBER OF FAILED FUEL RODS FOR THE BOUNDING FUEL HANDLING ACCIDENT IS GIVEN IN 15.7.6.3.

15.7.4.4 Barrier Performance

This failure occurs in the fuel handling building outside the normal barriers (RCPB and containment). Therefore, this section is not directly applicable. The transport of fission products to the environment is discussed in the next section.

15.7.4.5 Radiological Consequences

~~Three separate radiological analyses are provided for this accident~~:

a. The ~~first~~ ANALYSIS is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet ~~10 CFR 100 guidelines~~ REGULATORY GUIDE 1.183 DOSE CRITERIA. This analysis is referred to as the "design basis analysis," and is based on a 24 hour radioactive decay period of the fuel.

~~b. The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis," and is based on a 24 hour radioactive decay period of the fuel.~~

~~c. The third analysis is an assessment of offsite and Control Room dose consequences when irradiated fuel is being handled after seven days of radioactive decay has occurred. This analysis considers both conservative design basis and realistic source terms.~~ When comparing a fuel handling accident inside containment with a fuel handling accident in the Fuel Handling Building, the inside containment event would be bounding due to higher kinetic energy and the greater number of fuel pins damaged. ~~Both analyses make~~ The ~~equivalent~~ assumption IS MADE that the activity which escapes from the pool is released immediately and directly to the environment. Thus, for this analysis, refer to Section 15.7.6.4 on the bounding analyses for the fuel handling accident inside containment.

~~For all analyses,~~ The fission product inventory in the fuel rods assumed to be damaged is based on operation at 3,833 MWt.

~~15.7.4.5.1 Design Basis Analysis Assuming 24 Hour Radioactive Decay of the Fuel~~

The design basis analysis is based on ASSUMPTIONS AND METHODS CONTAINED IN ~~NRC Standard Review Plan 15.7.4 and~~ NRC Regulatory Guide 1.183 ~~1.25~~. ~~Specific values of parameters used in the evaluation are presented in Table 15.7-17~~.

INFORMATION ONLY

15.7.4.5.1.1 Fission Product Release from Fuel

REFER TO SECTION 15.7.6.4.1

~~The following conditions are assumed applicable for this event:~~

~~a. The fuel rod gap activity is assumed to consist of 10% of the total halogen and noble gas activity in the rods at the time of the accident, except for Kr-85 which is assumed to be 30%.~~

~~b. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be available for release.~~

~~c. It is conservatively assumed that 100 percent of the noble gas plenum activity and 1.0 percent of the halogen plenum activity in the damaged fuel rods is released from the spent fuel pool to the fuel handling building atmosphere.~~

~~Based on the above conditions the activity airborne in the fuel handling building is presented in Table 15.7-18.~~

15.7.4.5.1.2 Fission Product Transport to the Environment

REFER TO SECTION 15.7.6.4.1

~~In accordance with the criteria presented in Regulatory Guides 1.25 and 1.52 it is assumed that the airborne activity in the fuel handling building (Table 15.7-18) is released to the environment over a 2 hour period via a 95 percent iodine efficient FHAES. The total activity released to the environment is presented in Table 15.7~~-19.

15.7.4.5.1.3 Results

The calculated exposures for the ~~design basis analysis~~ BOUNDING FUEL HANDLING ACCIDENT are presented in Table 15.7-~~20~~ 35 and are ~~well~~ within the ~~guidelines of 10 CFR 100~~. DOSE CRITERIA OF REG GUIDE 1.183.

~~15.7.4.5.2 Realistic Analysis Assuming 24 Hour Radioactive Decay of the Fuel~~

~~The realistic analysis is based on a realistic but still conservative assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.7-17.~~

~~15.7.4.5.2.1 Fission Product Release from Fuel~~

~~Fission product release estimates for the fuel handling accident are based on the following assumptions:~~

~~a. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments.[3]~~

b. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

c. It is conservatively assumed that 100 percent of the noble gas plenum activity and 1.0 percent of the halogen plenum activity in the damaged fuel rods is released from the spent fuel pool to the fuel handling building atmosphere.

Based on the above conditions the activity airborne in the fuel handling building is presented in Table 15.7-21.

15.7.4.5.2.2 Fission Product Transport to the Environment

It is conservatively assumed that all activity released to the fuel handling building is released to the environment in the first two hours after the accident via a 95 percent iodine efficient FHAES. Based on these assumptions, the activity released to the environment is shown in Table 15.7-22.

15.7.4.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-23 and are well below the guidelines set forth in 10 CFR 100.

15.7.4.5.3 Design Basis and Realistie Analyses Assuming Seven Day Radioactive Decay

The radiological releases from a fuel handling accident inside containment (based on a seven day decay) are larger than those from a fuel handling accident outside containment (based on a seven day decay). Therefore, refer to Section 15.7.6.4.3 for details related to the bounding analyses for a fuel handling accident inside containment.

INFORMATION ONLY

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

This accident is not affected by the reload analysis.

15.7.5.1 Cask Drop from Transport Vehicle

In the unlikely event that the fuel cask falls from the transport vehicle, the maximum height which the cask will drop should be in general less than 10 ft. Since the cask is designed to withstand a 30 ft drop onto a non-yielding surface without failure, the fall from the transport vehicle will cause no failure of the cask.

15.7.5.2 Cask Drop from Crane

The Mark III containment design includes a separate fuel handling building. The spent fuel storage pools in this building are arranged so that the overhead crane which handles the cask cannot possibly move the cask above the spent fuel storage pool. This precludes the possibility of the cask falling on the stored spent fuel bundles. Also, the pools are arranged so that a rupture of the cask loading pool floor will not drain water from the spent fuel storage pool. The cask loading area design and operating procedures are specifically formulated so that a cask drop will not result in failure of the cask.

15.7.6 FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

The radiological exposures were recalculated ~~for Cycle 9~~ incorporating GE14 Fuel and power uprate analysis resulting in exposures ~~well below 10 CFR 100 Guidelines.~~ WITHIN THE LICENSE BASIS LIMITS OF 6.3 REM TEDE (OFFSITE) & 5 REM TEDE (CONTROL ROOM).

~~The analysis in Section 15.7.6.4.3 assumes a seven day decay time prior to the accident occurring. The seven day value forms the definition of "recently irradiated fuel" as identified in the Technical~~
THE ANALYSIS IS REVIEWED EACH CYCLE TO VERIFY THE INPUT ASSUMPTIONS & RESULTS REMAIN VALID.

INFORMATION ONLY

~~Specifications. This analysis is reviewed each cycle to verify that the seven day assumption is valid. As a result of this review, the definition of "recently irradiated fuel" may change to some value other than what is analyzed in this section~~.

15.7.6.1 Identification of Causes and Frequency Classification

15.7.6.1.1 Identification of Causes

Various mechanisms for fuel failure during refueling have been investigated. Procedural controls, backed up by the refueling interlocks, impose restrictions on the movement of refueling equipment and control rods, to prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system is able to initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned

INFORMATION ONLY

criticality tests with the reactor vessel head off. It is concluded that the accident that results in the release of significant quantities of fission products during this mode of operation with the greatest analyzed radiological consequences is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

The movement of non-fuel items over irradiated fuel inside containment will be administratively controlled such that the radiological consequences associated with their accidental dropping with or without primary or secondary containment will remain bounded by that of a dropped fuel bundle.

15.7.6.1.2 Frequency Classification

This event has been categorized as a limiting fault.

15.7.6.2 Sequence of Events and Systems Operation

15.7.6.2.1 Sequence of Events

The sequence of events which is assumed to occur is as follows:

	Event	Approximate Elapsed Time
a.	THE REFUELING Channeled fuel bundle being removed from ~~reactor vessel by crane~~. Fuel bundle is dropped from maximum height allowed by the refueling equipment. Fuel bundle strikes core.	0
b.	Some rods in both dropped and struck bundles fail releasing radioactive gases to pool water.	0
c.	Gases pass from water immediately to building.	0
d.	Containment vessel and drywell purge ventilation system isolates due to high radiation signal. NOT CREDITED IN THE DOSE ANALYSIS.	20 sec

15.7.6.2.1.1 Identification of Operator Actions

The accident analysis does not assume any operator actions for the mitigation of this event.

15.7.6.3 Core and System Performance

The methods used for this evaluation are the same as those presented in Section 15.7.4.3, FOR THE INITIAL CYCLE. THE ANALYSIS CREDITS 151 FAILED FUEL PINS BASED ON GE METHODOLOGY FOR GE 14 BUNDLE AND A TRIANGULAR FUEL HANDLING MAST.

15.7.6.4 Radiological Consequences

Three separate radiological ~~analyses~~ CASES are provided for this accident:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet ~~10 CFR 100 guidelines~~ REG GUIDE 1.183 DOSE CRITERIA. This analysis is ~~referred to as the "design basis analysis," and~~ is based on a 24 hour radioactive decay period of the fuel. THIS IS CONSIDERED THE BASE CASE AND TAKES NO CREDIT FOR ENGINEERED SAFETY FEATURES, ISOLATIONS, OR FILTRATION

INSERT #1

b. ~~The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis," and is based on a 24 hour radioactive decay period of the fuel~~.

INSERT #2

c. ~~The third analysis is an assessment of offsite and Control Room dose consequences when irradiated fuel is being handled after seven days of radioactive decay has occurred with the assumption that the release immediately and directly releases to the environment. This analysis considers both conservative design basis and realistic source terms.~~

INSERT #2A

For all ~~analyses~~ CASES, the fission product inventory in the fuel rods assumed to be damaged is based on operation at 3,833 MWt. Specific values for parameters used ~~in the first two analyses~~ are provided in Table 15.7-~~24~~32. ~~Specific values for the third analysis are provided in Table 15.7-32~~.

INSERT #2B

Insert #1 (15.7.6.4.b)
The second case is identical to the first case but was performed to determine the effect of control room isolation and fresh air intake. The control room dose was calculated assuming that once the available activity is introduced into the control room, the fresh air intake was isolated without any additional inleakage. At two hours, outside air purge is assumed to initiate and continue for the remainder of the 30-day dose analysis.

Insert #2 (15.7.6.4.c)
The third case is also identical to the first case but was performed to determine the effect of control room isolation and emergency recirculation filtering. The control room dose was calculated assuming that once the available activity is introduced into the control room, the fresh air intake was isolated without any additional inleakage. At two hours, the control room emergency recirculation system was assumed to initiate and continue for the remainder of the 30-day dose analysis. A filtration efficiency for the charcoal of 50% was assumed in order to be consistent with Table 15.6-14.

Insert #2A
Note. The second and third cases were performed to examine the flexibility the Control Room operators have in using ventilation to ensure that there were no dose outliers. The results show that even if the operators take 2 hours to initiate an action, (i.e., re-initiate normal intake or utilize recirculation filtration) the doses remain below the license basis limits.

Insert #2B
An additional scenario was reviewed in the event a fuel bundle was dropped when travelling through the refueling shield inside containment. This reduces the amount of water coverage resulting in a lower Decontamination Factor. The resultant dose is less than the cases described above when considering that the damage is limited to the number of pins in one fuel bundle.

15.7.6.4.1 Design Basis Analysis Assuming 24 Hour Radioactive Decay of the Fuel

a. Fission Product Release from Fuel

INSERT #3

~~The fission product activity released from the fuel damaged as~~ a ~~result of a fuel handling accident is calculated using the methods outlined in Section 15.7.4.5.1.1. A total of 124 fuel rods fail as a result of this accident. For the initial cycle, a total of 151 fuel rods fail given a core loaded with GE14 fuel~~.

Insert 3 (15.7.6.4.1.a)

The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods below:

1) The fuel rod gap activity is assumed to consist of 5% of the total halogen and noble gas activity in the rods at the time of the accident, except for KR-85, which is assumed to be 10% and I-131 which is assumed to be 8%.
2) Twelve percent of the alkali metals are available for release but are retained in the water.

A total of 151 fuel rods fail as a result of the accident given a core loaded with GE14 fuel.

b. Fission Product Activity ~~Airborne in the Reactor Building~~ RELEASED

The following assumptions and initial conditions are used in calculating the fission product activity released to the ~~reactor building~~. ENVIRONS.

1. The iodine gap inventory is composed of ~~inorganic~~ ELEMENTAL species (99.~~75~~ 85 percent) and organic species (0.~~25~~ 15 percent).

2. The minimum water depth between the top of the damaged fuel rods and the containment pool surface is 23 feet.

3. The pool decontamination factors for the inorganic and organic species of iodine are ~~133~~ 500 and 1, respectively, giving an overall effective decontamination factor of ~~100~~ 200 (i.e., 99.5 ~~99~~ percent of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for ~~inorganic~~ ELEMENTAL and organic iodine species results in the iodine above the fuel pool being composed of ~~75~~ 57 percent ~~inorganic~~ ELEMENTAL and ~~25~~ 43 percent organic species.

4. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1). PARTICULATE RADIONUCLIDES ARE ASSUMED TO BE RETAINED BY THE WATER

5. The effects of plateout and fallout are neglected.

6. THE ACTIVITIES WITHIN THE FAILED RODS ARE ASSUMED TO HAVE BEEN DECAYED 24 HOURS PRIOR TO THE ACCIDENT.

7. ALL ACTIVITY RELEASED FROM THE POOL TO THE CONTAINMENT IS RELEASED DIRECTLY TO THE ENVIRONMENT (i.e., FILTERING & CONTAINMENT INTEGRITY IS NOT CREDITED)

Based on these assumptions, the activity released from the pool to the ~~reactor building~~ ENVIRONMENT is listed in Table 15.7-~~25~~ 34.

c. ~~Fission Product Release to Environs.~~

The following assumptions and initial conditions are used in calculating the fission products released to the environs:

1. The containment vessel and drywell purge system are in operation at the time the accident occurs. These systems are described in Section 9.4. It is conservatively assumed that isolation of the containment vessel and drywell purge system will occur 20 seconds after the release of fission product activity from the containment pool due to a high radiation signal in this system.

2. All activity during the first 20 seconds after the accident is assumed to be released to the environs via the containment vessel and drywell purge exhaust system filter as a "puff" release.

3. No credit is taken for filtering iodine during the first 20 seconds after the accident.

4. The activity remaining in containment is released to the environs (via the annulus exhaust gas treatment system).

Based on these assumptions, the activity released to the ~~environment is presented in Table 15.7-26.~~

d. Results

Based on these assumptions, the TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE) ~~integrated whole body doses and integrated thyroid dose~~ at the exclusion boundary, ~~and~~ low population zone, AND CONTROL ROOM are summarized in Table 15.7-35 ~~27~~. The doses at these distances are ~~well below the 10 CFR 100 limits~~, WITHIN THE LICENSING BASIS LIMITS OF 6.3 REM (TEDE) (OFFSITE) & 5 REM (TEDE) (CONTROL ROOM)

~~15.7.6.4.2 Realistic Analysis Assuming 24 Hour Radioactive Decay of the Fuel~~

The remainder of this section is removed. Next page is 15.7-41b.

a. Fission Product Release from Fuel

The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods outlined in Section 15.7.4.5.2.1. As a result of this accident, 124 fuel rods are assumed to fail for the initial cycle. A total of 151 fuel rods are assumed to fail for a core loaded with GE14 fuel.

b. Fission Product Activity Released to Containment

The following assumptions and initial conditions are used in calculating the fission products released to the containment:

1. The fission product activity released to the containment will be inversely proportional to the removal efficiency of the water in the upper containment pool. Because water has a negligible effect on removal of the noble gases, noble gases are assumed to be instantaneously released from the pool to the containment.

2. The iodine activity in the fuel rod plena is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

(next page is 15.7-41b)

~~Based on these assumptions, the activity released to the containment~~ pool and to the environment is given in Table 15.7-33 and Table 15.7-34 respectively.

c. Results

Based on these assumptions, the integrated whole body doses and integrated thyroid doses at the exclusion boundary, low population zone, and Control Room are summarized in Table 15.7-35. The doses at these distances are well below the 10CFR100 limits, and within the General Design Criteria 19 limit for the Control Room.

15.7.7 REFERENCES FOR SECTION 15.7

1. Nguyen, D., "Realistic Accident Analysis - The RELAC Code," October 1977, (NEDO-21142).

2. Bunch, F. D., "Dose to Various Body Organs from Inhalation or Ingestion of Soluble Radionuclides," IDO-12054, AEC Research and Development Report, TID-4500, August 1966.

3. N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," March 1969, (APED 5756).

4. General Electric Company "General Electric Standard Application for Reactor Fuel," including the United States Supplement, NEDE-24011-P-A and NEDE-P-A-US (latest approved revision).

5. Reg. Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the purpose of evaluating Compliance with 10CFR Part 50, Appendix I, Rev. 1, October 1977.

6. REGULATORY GUIDE 1.183, ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS, JULY 2000

Tables 15.7-17 through 15.7-30 are deleted.

TABLE 15.7-17

FUEL ~~HANDLING ACCIDENT OUTSIDE CONTAINMENT PARAMETERS~~
TABULATED FOR POSTULATED ACCIDENT ANALYSIS

INFORMATION ONLY

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3,833 MWt	3,833 MWt
B. Radial peaking factor	1.7	1.6
C. Fuel damage (GE14)	151 rods	151 rods
D. Release of activity by nuclide	Section 15.7.4.5.1.1	Section 15.7.4.5.2.1
E. Iodine fractions		
(1) Organic	0.0025	0.0025
(2) Elemental	0.9975	0.9975
(3) Particulate	0	0
II. Data and assumptions used to estimate activity released		
A. Fuel handling building leak rate	100%/2 hr	100%/2 hr
B. Adsorption and filtration efficiencies		
(1) Organic iodine	95%	95%
(2) Elemental iodine	95%	95%
(3) Particulate iodine	95%	95%
C. All other pertinent data and assumptions	None	None

TABLE 15.7-32

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS
(ASSUMING ~~7 DAY~~ 24 HOUR RADIOACTIVE DECAY)

INFORMATION ONLY

		Design Basis Assumptions	~~Realistic Case Assumptions~~
I.	Data and assumptions used to estimate radioactive source from postulated accidents		
A.	Power level	3,833 MWt	3,833 MWt
B.	~~Burn-up~~ INSERT #6	~~Section 15.7.6.4~~	Section 15.7.6.4
C.	Fuel damage (GE14)	151 rods	151 rods
D.	Release of activity to containment pool by nuclide, per failed rod	Section 15.7.6.4.~~3~~1	Section 15.7.6.4.3
E.	Iodine fractions - organic	.~~0025~~ .0015	.0025
	- ~~inorganic~~ ELEMENTAL	.~~9975~~ .9985	.9975
F.	RADIAL PEAKING FACTOR	2.0	
II.	Data and assumptions used to estimate activity released		
A.	Primary containment leak rate	Instantaneous total release of all activity leaving pool to environment	Instantaneous total release of all activity leaving pool to environment
B.	Secondary containment leak rate	N/A	N/A
C.	Isolation valve closure times	N/A	N/A
D.	Filtration efficiencies	N/A	N/A
E.	Recirculation systems parameters (flow rates vs. time, mixing factor, etc.)	N/A	N/A

Insert #6 (Table 15.7-32, Item I.B)

B. Burn-up	-Peak rod exposure is less than 62,000MWD/MT -Between 54,000 and 62,000MWD/MT, the maximum linear heat generation rate does not exceed 6.3 KW/Ft peak rod average exposure.

INFORMATION ONLY

TABLE 15.7-32 (Continued)

INFORMATION ONLY

		Design Basis Assumptions	~~Realistic Case Assumptions~~
F.	Containment spray parameters (flow rate, drop size, etc.)	N/A	~~N/A~~
G.	Containment volumes	N/A	~~N/A~~
H.	Pool Removal (iodine partition factor)	~~100~~ 200	~~100~~
I.	All other pertinent data and assumptions	Section 15.7.6	~~Section 15.7.6~~
J.	Activity released to environment	Table 15.7-~~33~~34	~~Table 15.7-33~~
K.	Control Room Parameters	See below	~~N/A~~
	1. Volume (ft^3)	367,070 ft^3	
INSERT #4	~~2. Design Flow (cfm)~~	~~6,600 (Instantaneous air mixing at + 10% of design flow)~~	
	~~3. Filter Efficiency~~	~~0 (No filtered recirculation assumed)~~	
III. Dispersion Data			
A.	Boundary and LPZ distances (m)	863/4002	~~863/4002~~
B.	Offsite X/Q's (Corresponding to 7 year meteorological data for 0-2 hr for SB/0-8 hr for LPZ)	4.3E-4/4.8E-5	~~4.3E-4/4.8E-5~~
C.	Control Room X/Q's (0-8 hr)	3.5E-4 ~~See below~~	~~N/A~~
	~~(1) 0-8 hrs~~	~~3.5E-4~~	
	~~(2) 8-24 hrs~~	~~2.1E-4~~	
	~~(3) 1-4 days~~	~~1.1E-4~~	
	~~(4) 4-30 days~~	~~5.75E-5~~	
IV. Dose Data			
A.	Method of dose calculation	Section 15.0.3.5	~~Section 15.0.3.5~~

Insert #4 (Table 15.7-32 Item II.K)

2. Design Flow (cfm)	Intake	Exhaust (AIR PURGE)	Emergency Recirculation
Case 1	6600 (Normal +10%)	5400 (Normal –10%)	0
Case 2	6600 Isolated after activity introduced into Control Room	5400 Started after 2 hrs	0
Case 3	6600 Isolated after activity introduced into Control Room	0	27,000 (Normal –10%) Started after 2 hrs Filter efficiency = 50%

TABLE 15.7-32 (Continued)

INFORMATION ONLY

		Design Basis Assumptions	~~Realistic Case Assumptions~~
B.	Dose conversion assumptions	~~Section 15.0.3.~~5	~~Section 15.0.3.5~~
1.	Dose conversion assumptions (Offsite)	Section 15.0.3.5	
2.	Dose Conversion Assumptions (Control Room)	Section 15.0.3.5	
		~~International Commission of Radiological Protection (ICRP) 30 Dose Conversion Factors (rem/Ci)~~	~~N/A~~

~~Isotope~~	
~~I-131~~	~~1.10E6~~
~~I-132~~	~~6.30E3~~
~~I-133~~	~~1.80E5~~
~~I-134~~	~~1.10E3~~
~~I-135~~	~~3.10E4~~

C.	Peak activity concentrations in containment	N/A	~~N/A~~
D.	Doses	Table 15.7-35	~~Table 15.7-3~~5

INFORMATION ONLY

TABLE 15.7-33

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING 7 DAY RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS AND REALISTIC ANALYSIS
ACTIVITY RELEASED TO THE CONTAINMENT POOL (CURIES)*

Isotope	Design Basis Source Terms Activity	Realistic Source Terms Activity
I-131	2.1E+4	1.7E+3
I-132	**	**
I-133	2.0E+2	5.1E+0
I-134	**	**
I-135	2.0E-3	2.9E-5
Kr-83m	**	**
Kr-85m	7.8E-8	4.3E-9
Kr-85	9.2E+2	6.2E+2
Kr-87	**	**
Kr-88	**	**
Kr-89	**	**
Xe-131m	1.9E+2	4.8E+1
Xe-133m	1.4E+3	1.4E+2
Xe-133	2.6E+4	4.3E+3
Xe-135m	**	**
Xe-135	2.8E-1	5.8E-2
Xe-137	**	**
Xe-138	**	**

* GE14 Fuel @ 3,833 MWt

** Negligible levels of activity

INFORMATION ONLY

TABLE 15.7-34

24-HOUR

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING ~~7 DAY~~ RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS ~~AND REALISTIC~~ ANALYSIS
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)*

I-129 6.27E-6
I-130 1.37E 0

Isotope	Design Basis Source Terms Activity	Realistic Source Terms Activity
I-131	1.82E+2 ~~2.1E+2~~	1.7E+1
I-132	1.45E+2 ~~**~~	**
I-133	1.14E+2 ~~2.0E+0~~	5.1E-2
I-134	6.00E-6 ~~**~~	**
I-135	1.88E+1 ~~2.0E-5~~	2.9E-7
Kr-83m	1.15E+1 ~~**~~	**
Kr-85	7.49E+2 ~~9.2E+2~~	6.2E+2
Kr-85m	1.49E+2 ~~7.8E-8~~	4.3E-9
Kr-87	2.42E-2 ~~**~~	**
Kr-88	4.61E-1 ~~**~~	**
~~Kr-89~~	~~**~~	~~**~~
Xe-129m	2.03E-1	
Xe-131m	2.74E+2 ~~1.9E+2~~	4.8E+1
Xe-133m	1.40E+3 ~~1.4E+3~~	1.4E+2
Xe-133	4.60E+4 ~~2.6E+4~~	4.3E+3
Xe-135m	6.00E+2 ~~**~~	**
Xe-135	1.27E+4 ~~2.8E-1~~	5.8E-2
~~Xe-137~~	~~**~~	~~**~~
~~Xe-138~~	~~**~~	~~**~~

* GE14 Fuel @ 3,833 MWt

** ~~Negligible levels of activity~~ RETAINED COMPLETELY IN THE POOL

Br 82	5.58E-2
Br 83	1.49E-2
Rb 87	0 **
Rb 88	0 **
~~Rb~~	
Cs 135	0 **

Revision 11
September, 2001

INFORMATION ONLY

TABLE 15.7-35

24-HOUR

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING ~~7-DAY~~ RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS ~~AND REALISTIC~~ ANALYSIS
RADIOLOGICAL EFFECTS

~~DESIGN BASIS SOURCE TERMS~~

	Whole Body Dose (rem)	Inhalation Dose (rem)	Beta Skin Dose (rem)
Exclusion Area (863 Meters)	1.17E-1	4.71E+1	N/A
Low Population Zone (4,002 Meters)	1.40E-2	5.25	N/A
Control Room	1.27E-2	2.76E+1	3.37E-1

REALISTIC SOURCE TERMS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area (863 Meters)	1.87E-2	3.85
Low Population Zone (4,002 Meters)	2.10E-3	0.43
Control Room	N/A	N/A

NOTE: The above radiological effects have been updated to reflect the scaled increases associated with Power Uprate to 3,758 MWt.

INSERT #5

DESIGN BASIS SOURCE TERM

	TEDE Dose (rem)	Licensing Limit (rem)
Exclusion Area (863 Meters)	1.44	6.3
Low Population Zone (4002 Meters)	0.161	6.3
Control Room:		
Case 1	1.03	5
Case 2	2.81	5
Case 3	2.97	5

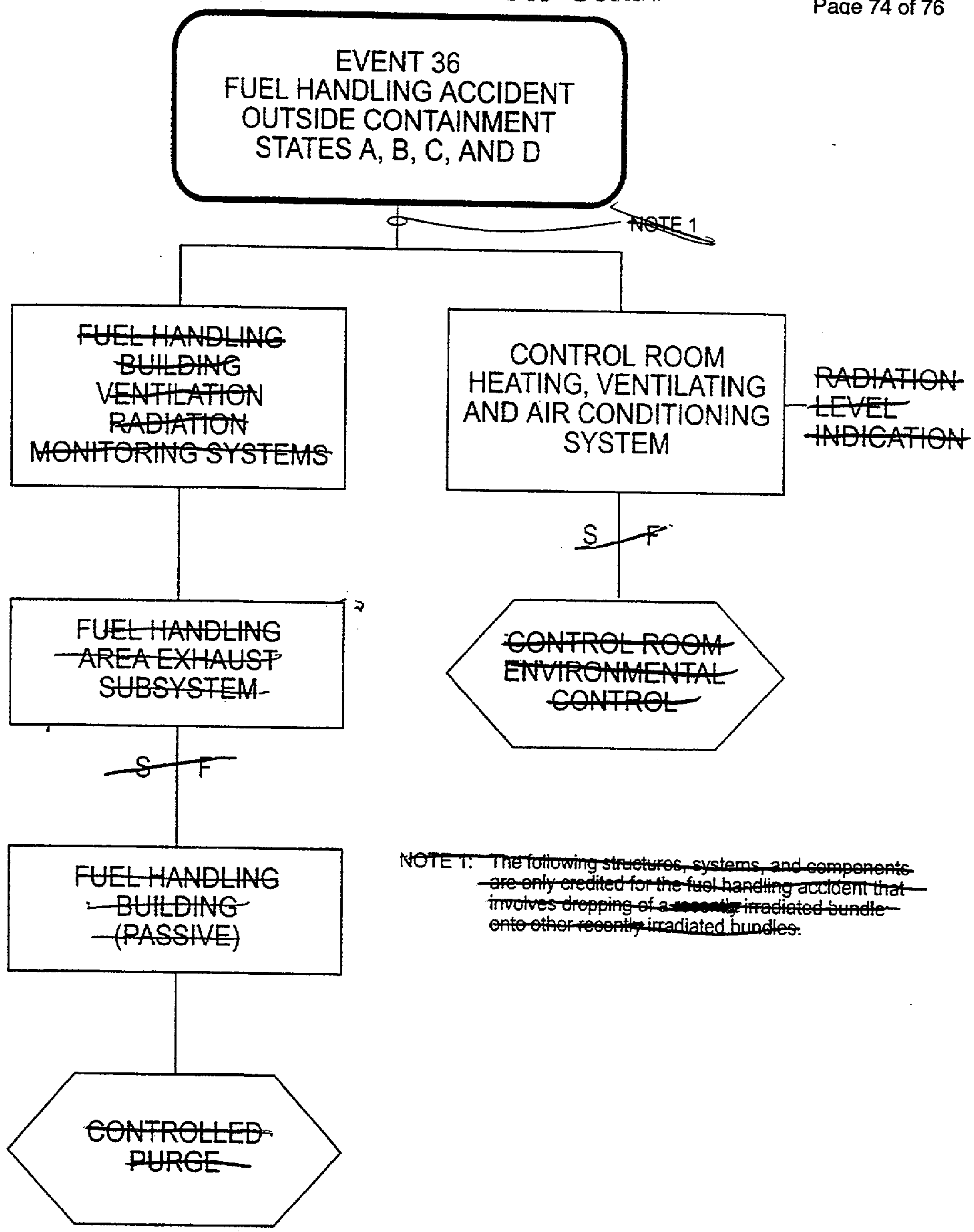
Figure 15A.6-35 presents the different protection sequences for the control rod drop accident. As shown in Figure 15A.6-35, the reactor is automatically scrammed and isolated. For all design basis cases, the neutron monitoring, reactor protection and control rod drive systems will provide a scram from high neutron flux. After the reactor has been scrammed, core cooling is accomplished by either the RCIC or the HPCS or the normal feedwater system.

b. Event 36 - Fuel Handling Accident Outside Containment

Because a fuel-handling accident can potentially occur any time when fuel assemblies are being manipulated in the fuel handling building, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A.6-36. It is important to note that the systems, structures, and components described within Figure 15A.6-36 are only credited for the fuel handling accident that involves dropping of a ~~recently~~ irradiated bundle onto other ~~recently~~ irradiated bundles.

It is important to note that the systems, structures, and components described within Figure 15A.6-43 are only credited for the fuel handling accident that involves dropping of an ~~recently~~ irradiated bundle onto other ~~recently~~ irradiated bundles.

INFORMATION ONLY

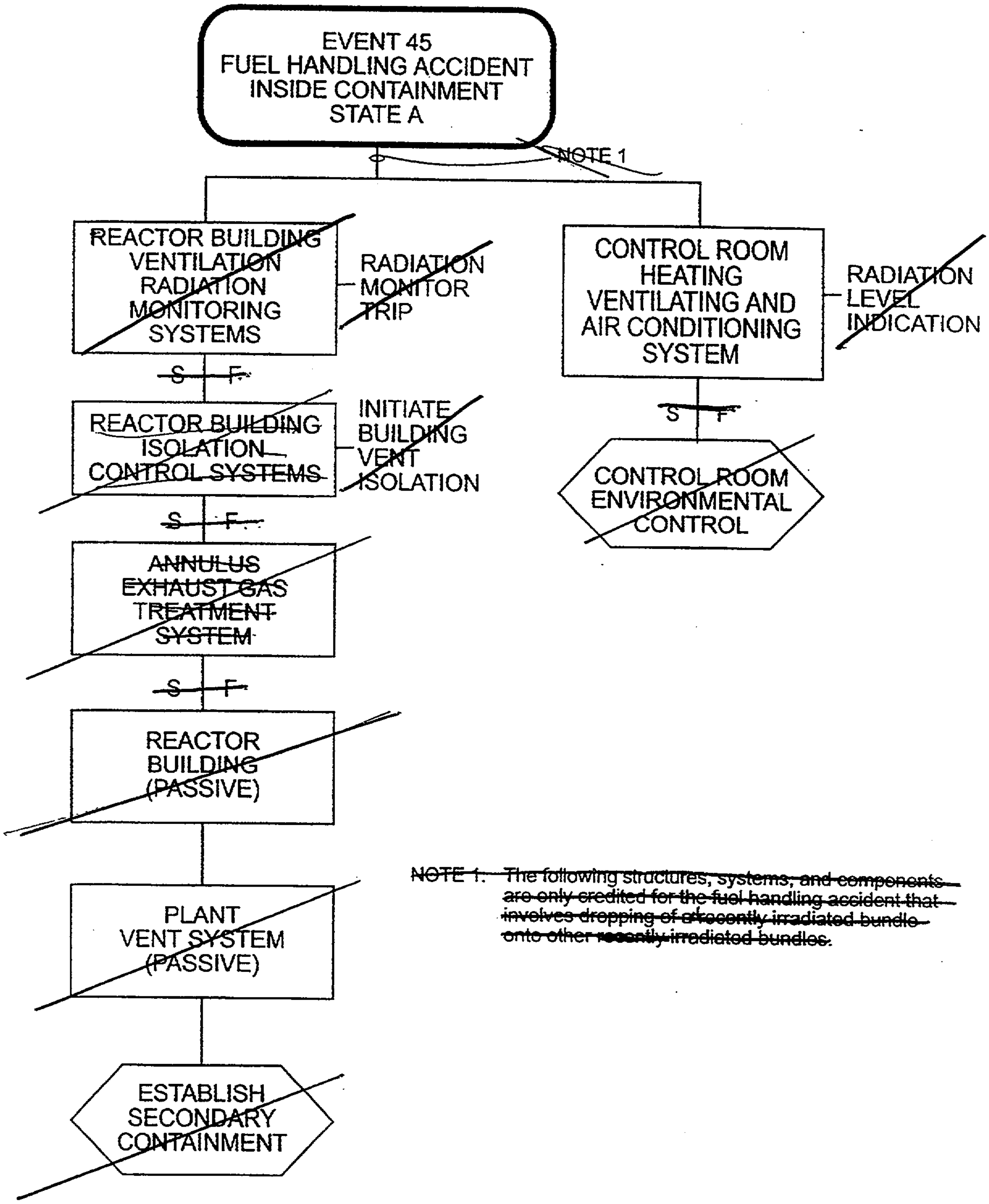


(Rev. 10 10/99)

Protective Sequences for Fuel Handling Accident Outside Containment

Figure 15A.6-36

INFORMATION ONLY



(Rev. 10 10/99)

Protective Sequences for Fuel Handling Accident Inside Containment

Figure 15A.6-43

TABLE 15A.2-4

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY: DESIGN BASIS ACCIDENTS

Unacceptable Consequences

4-1 Radioactive material release exceeding the guideline values of 10 CFR 100 (for the design-basis RAST LOCA analysis, the licensing basis offsite dose limit is 25 rem TEDE, FOR THE DESIGN BASIS FUEL HANDLING ACCIDENT, THE LICENSING BASIS OFFSITE DOSE LIMIT IS 6.3 REM TEDE.)

4-2(1) Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.

4-3 Nuclear system stresses exceeding that allowed for accidents by applicable industry codes.

4-4 Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.

4-5 Overexposure to radiation of plant main control room personnel.

NOTE:

1. Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).