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U-603485
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July 5, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

10CFR50.90

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Request for Amendment to Technical Specifications that Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term

References:
(1) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000.
(2) Letter from D. Hood (U. S. NRC) to G. Van Middlesworth (Nuclear Management Company, LLC), "Duane Arnold Energy Center – Issuance of Amendment Regarding Secondary Containment Operability During Movement of Irradiated Fuel and Core Alterations (TAC No. MB1569)," dated April 6, 2001.
(3) Letter from S. Sekerak (U. S. NRC) to W. Eaton (Entergy Operations), "Grand Gulf Nuclear Station (GGNS), Unit 1 – Issuance of Amendment Re: Full-Scope Implementation of an Alternative Accident Source Term (TAC No. MA8065)," dated March 14, 2001.
(4) Technical Specifications Task Force (TSTF) Traveler TSTF-51, "Revise containment requirements during handling of irradiated fuel and core alterations," Revision 2, dated October 1999.

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (i.e., AmerGen), proposes changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed TS changes revise the requirements that apply during the movement of irradiated fuel and during Core Alterations. Specifically, the proposed changes relax operability requirements for primary containment systems, secondary containment systems, and the Standby Gas Treatment (SGT) system.

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These proposed changes are based on a revised analysis of the Fuel Handling Accident (FHA) for CPS that conforms to the guidance in Reference (1). This selective implementation of the guidance in Reference (1) provides a new design and licensing basis for the FHA. The proposed changes are similar to the changes approved for the Duane Arnold and Grand Gulf stations as documented in References (2) and (3). In addition, the proposed changes are consistent with TSTF-51, Revision 2 (Reference 4), submitted to the NRC by the Technical Specifications Task Force (TSTF). Therefore, we request approval of these changes by January 20, 2002 with a 60-day implementation period. Approval by this date will support implementation during the next refueling outage scheduled to start on March 22, 2002.

This request is subdivided as follows.

1. Attachment A provides a description and safety analysis of the proposed changes.

2. Attachment B includes the marked-up TS pages with the proposed changes indicated. A marked-up copy of the affected TS Bases pages is provided for information only.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (a)(1) which provides information supporting a finding of no significant hazards consideration in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (c).

4. Attachment D provides information supporting an Environmental Assessment.

The proposed changes have been reviewed by the CPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board.

AmerGen is notifying the State of Illinois of this request for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Mr. R. W. Chickering at 217-937-3334.

Respectfully,

J. M. Heffley
Vice President
Clinton Power Station

RWC/blf

Attachments:

Affidavit
Attachment A: Description and Safety Analysis for the Proposed Changes
Attachment B: Marked-up TS Pages for the Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting An Environmental Assessment

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DEWITT	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	Docket Number
CLINTON POWER STATION, UNIT 1	)	50-461

SUBJECT: **Request for Amendment to Technical Specifications that Revise Plant System Requirements During Fuel Handling Based on Alternative Source Term**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

J. M. Heffley
Vice President
Clinton Power Station

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 5th day of

July, 2001.

Jacqueline S. Matthias
Notary Public

* OFFICIAL SEAL *
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Expires 11/24/2001

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGES

A. SUMMARY OF THE PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (i.e., AmerGen), proposes changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed changes revise various TS requirements that apply during the movement of irradiated fuel and during Core Alterations. Core Alterations is defined in the TS as the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Core Alterations do not include the movement of source range monitors, local power range monitors, intermediate range monitors, special movable detectors or control rod movement, provided there are no fuel assemblies in the associated core cell. Specifically, the proposed changes relax operability requirements for primary containment systems, secondary containment systems, and the Standby Gas Treatment (SGT) System. These proposed changes are based on the selective implementation of the alternative source term (AST) for the Fuel Handling Accident (FHA) as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 1), dated July 2000.

Additionally, CPS will administratively implement provisions of Section 11, "Assessment of Risk Resulting From Performance of Maintenance Activities," subsection 3.6.5 of Nuclear Utility Management and Resources Council (NUMARC) document NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3 (Reference 2), pertaining to the capability to restore secondary containment.

The proposed changes are described in Section E of this Attachment. The marked-up TS pages are shown in Attachment B. In addition, marked–up TS Bases pages reflecting these changes are provided for information only in Attachment B.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

B.1 TS Section 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"

TS Section 3.3.6.1, Table 3.3.6.1-1 footnote d requires that the instrumentation for each of the following functions be operable during Core Alterations and during movement of irradiated fuel assemblies in primary or secondary containment.

Function 2, "Primary Containment and Drywell Isolation"

- Item g, "Containment Building Fuel Transfer Pool Ventilation Plenum Radiation – High"
- Item h, "Containment Building Exhaust Radiation – High"
- Item i, "Containment Building Continuous Containment Purge (CCP) Exhaust Radiation – High"
- Item l, "Manual Initiation"

Function 4, "Reactor Water Cleanup (RWCU) System Isolation"

- Item h, "Manual Initiation"

TS Section 3.3.6.1 Condition N requires that if an inoperable channel is not restored to operable status or placed in trip within the allowed completion time, the associated penetration flow path should be isolated. Alternatively, the condition requires that the plant be placed in a condition where the TS Limiting Condition for Operation (LCO) does not apply. Therefore, Condition N allows for suspending Core Alterations and suspending movement of irradiated fuel assemblies in primary and secondary containment in lieu of isolating the affected penetration flow path.

B.2 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

TS Section 3.3.6.2, Table 3.3.6.2-1 footnote b requires that the instrumentation for the following functions be operable during Core Alterations and during movement of irradiated fuel assemblies in the primary or secondary containment (i.e., fuel building).

Function 3, "Containment Building Fuel Transfer Pool Ventilation Plenum Exhaust Radiation – High"
Function 4, "Containment Building Exhaust radiation – High"
Function 5, "Containment Building Continuous Containment Purge (CCP) Exhaust Radiation – High"
Function 7, "Manual Initiation"

TS Section 3.3.6.2, Table 3.3.6.2-1 footnote c requires that the instrumentation for the following function be operable during movement of irradiated fuel assemblies in the Fuel Building.

Function 6, "Fuel Building Exhaust Radiation – High"

B.3 TS Section 3.6.1.2, "Primary Containment Air Locks"

TS Section 3.6.1.2 requires that the primary containment air locks shall be operable during Core Alterations and during movement of irradiated fuel assemblies in primary or secondary containment.

B.4 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

TS Section 3.6.1.3 requires that the PCIVs shall be operable during Core Alterations and during movement of irradiated fuel assemblies in primary or secondary containment.

B.5 TS Section 3.6.4.1, "Secondary Containment"

TS Section 3.6.4.1 requires that the secondary containment shall be operable during Core Alterations and during movement of irradiated fuel assemblies in primary or secondary containment.

B.6 TS Section 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)"

TS Section 3.6.4.2 requires that the subject dampers be operable during Core Alterations and during movement of irradiated fuel assemblies in primary or secondary containment.

B.7 TS Section 3.6.4.3, "Standby Gas Treatment (SGT) System"

TS Section 3.6.4.3 requires that the SGT system shall be operable during Core Alterations and during movement of irradiated fuel assemblies in primary or secondary containment.

C. BASES FOR THE CURRENT REQUIREMENTS

C.1 TS Section 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"

The radiation monitoring instrumentation required to be operable by TS Section 3.3.6.1, Table 3.3.6.1-1, subfunctions 2.g, 2.h, and 2.i, initiate primary containment and drywell isolation as a result of high radiation levels that are an indication of gross fuel cladding failure. When a high radiation level is detected, valves in penetrations that communicate with the primary containment atmosphere are isolated to limit the release of fission products. Additionally, a high radiation level is assumed to initiate isolation of the penetrations that bypass secondary containment during a FHA (Reference 7). In addition, these functions provide an isolation signal to the drywell isolation valves. The closure 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 pressure suppression function of the drywell. These functions are required to be operable during Core Alterations and during movement of irradiated fuel in primary or secondary containment because detecting radioactive material releases due to fuel failures from dropped fuel assemblies must be provided to ensure offsite dose limits are not exceeded.

The manual initiation function required by TS Section 3.3.6.1, Table 3.3.6.1-1, subfunction 2.l introduces a signal into the primary containment and drywell isolation logic that is redundant to the automatic protective instrumentation and provides manual isolation capability. This function initiates closure of valves that isolate primary containment penetrations which bypass secondary containment. Thus, this function is also required under those conditions in which secondary containment is required to be operable. Similarly, the manual initiation function required by TS Section 3.3.6.1, Table 3.3.6.1-1, subfunction 4.h introduces a signal into the RWCU System isolation logic that is redundant to the automatic protective instrumentation and provides manual isolation capability. This function also initiates closure of valves that isolate primary containment penetrations which bypass secondary containment and therefore is required under those conditions in which secondary containment is required to be operable.

TS Section 3.3.6.1 Condition N requires isolation of the affected penetration flow path in the event an instrumentation channel is not returned to operable status or placed in trip within the allowed Completion Time. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable instrumentation. Rather

than isolating the affected penetration flow path, Condition N allows for placing the plant in a condition in which the LCO does not apply. Therefore, if applicable, Core Alterations and movement of irradiated fuel assemblies must be immediately suspended.

C.2 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

High radiation levels in the primary or secondary containment (i.e., fuel building) exhaust are an indication of possible gross failure of the fuel cladding. The release may originate from a break in the reactor coolant pressure boundary (RCPB) or from a FHA in containment or the fuel building. The radiation monitoring instrumentation required to be operable by TS Section 3.3.6.2, Table 3.3.6.2-1, Functions 3, 4, 5, and 6 initiate secondary containment isolations and actuation of the SGT System to limit the release of fission products as assumed in the Updated Safety Analysis Report (USAR) safety analysis. These functions are required to be operable during Core Alterations and during movement of irradiated fuel in the primary containment or the fuel building, as applicable, because the capability to detect fuel failures due to fuel failures from dropped fuel assemblies must be provided to ensure that the offsite dose limits are not exceeded.

The manual initiation function required by TS Section 3.3.6.2, Table 3.3.6.2-1, Function 7 introduces a signal into the secondary containment isolation logic that is redundant to the automatic protective instrumentation channels and provides manual isolation capability. This function is required to be operable during Core Alterations and movement of irradiated fuel assemblies in the primary containment or the fuel building since these are the conditions in which the secondary containment isolation automatic functions are required to be operable.

C.3 TS Section 3.6.1.2, "Primary Containment Air Locks"

The primary containment air locks required by TS Section 3.6.1.2 function to provide personnel access to the primary containment and to provide containment isolation during the process of personnel entry and exit. The air locks are designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment. As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through the range of transients and accidents up to and including postulated design basis accidents such as the FHA and the Loss of Coolant Accident (LOCA).

The upper containment personnel airlock is also credited in the analysis of the FHA. The upper containment personnel air lock performs no active function in response to the postulated accident; however, its leak tightness is required to ensure that the release of radioactive materials from primary containment is restricted to those leakage paths assumed in the accident analysis, and the fission products released by the FHA will be treated by the SGT System.

C.4 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The PCIVs in combination with other accident mitigation systems, are necessary to limit fission product release during and following postulated design basis accidents. Primary containment isolation within the time limits specified for those PCIVs

designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a design basis accident, including the FHA. Specifically, in the event of an FHA, PCIVs isolate lines that bypass secondary containment to control the release of radioactive material.

C.5 TS Section 3.6.4.1, "Secondary Containment"

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. In conjunction with operation of the SGT System and closure of certain valves whose lines penetrate the secondary containment, 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 (e.g., during Core Alterations or during movement of irradiated fuel assemblies in primary or secondary containment), when primary containment is not required to be operable, or that take place outside primary containment. Fission products in the secondary containment will be treated by the SGT System prior to release to the environment. The leak tightness of the secondary containment 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.

C.6 TS Section 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)"

The SCIDs, in combination with other accident mitigation systems, function to limit fission product release during and following postulated design basis accidents. The boundary established by the SCIDs is required to ensure that leakage from the primary containment is processed by the SGT System before being released to the environment. Maintaining SCIDs operable with isolation times within limits ensures that fission products will remained trapped inside of secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

C.7 TS Section 3.6.4.3, "Standby Gas Treatment (SGT) System"

The SGT System functions to ensure that radioactive materials that leak from primary containment into secondary containment and radioactive materials that are released directly into secondary containment following a design basis accident, including the FHA, are filtered and adsorbed prior to exhausting to the environment. The SGT System is required by 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 41, "Containment atmosphere cleanup."

D. NEED FOR REVISION OF THE REQUIREMENTS

The proposed changes to the TS will allow the plant to apply the results of the plant-specific AST analysis while using the regulatory guidance in Reference (1) and meeting the requirements in Reference (3). The proposed changes will shorten refueling outages by facilitating passage through doors and hatches and by facilitating the scheduling of maintenance activities without increased consequences or risk to the public and control room operators. This application of the methods in Reference 1 results in CPS specific TS

changes that are comparable to TS changes approved by the NRC for the Duane Arnold and Grand Gulf stations in References (4) and (5). In addition, these changes are consistent with Technical Specifications Task Force (TSTF) Traveler 51, Revision 2 (Reference 6).

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed TS changes are as follows.

E.1 TS Section 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"

Delete footnote d, "During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment," on TS Table 3.3.6.1-1, "Primary Containment and Drywell Isolation Instrumentation." Also remove the corresponding footnote d designation on TS Table 3.3.6.1-1, subfunctions 2g, 2h, 2 i, 2l, and 4h. Re-designate the remaining footnote letters and the footnote letters on the other functions on TS Table 3.3.6.1-1 in order to account for the removal of footnote d. These changes remove the requirement for radiation monitoring functions and manual isolation initiation functions to be operable during Core Alterations and during movement of irradiated fuel assemblies in the primary or secondary containment.

Delete Required Action N.2.1, "Suspend CORE ALTERATIONS," Required Action N.2.2, "Suspend movement of irradiated fuel assemblies in primary and secondary containment," and the associated completion times from TS Section 3.3.6.1 Condition N. This change removes the requirement to suspend Core Alterations and suspend movement of irradiated fuel assemblies in primary and secondary containment under these conditions.

E.2 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

Delete footnote b, "During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment," and footnote c, "During movement of irradiated fuel assemblies in the fuel building," on TS Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation." Also remove the corresponding footnote b and c designation on TS Table 3.3.6.2-1, Functions 3, 4, 5, 6, and 7. These changes remove the requirement for radiation monitoring functions and manual isolation initiation functions to be operable during Core Alterations and during movement of irradiated fuel assemblies in the primary or secondary containment.

E.3 TS Section 3.6.1.2, "Primary Containment Air Locks"

Delete "During CORE ALTERATIONS," and "During movement of irradiated fuel assemblies in the primary or secondary containment," from the Applicability statement for TS LCO 3.6.1.2. In addition, delete "movement of irradiated fuel assemblies in the primary or secondary containment, CORE ALTERATIONS, or" from TS Section 3.6.1.2 Condition E. Also, delete the Note, "LCO 3.0.3 is not applicable," Required Actions E.1 and E.2, and associated completion times from Condition E.

E.4 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

Delete "During CORE ALTERATIONS," and "During movement of irradiated fuel assemblies in the primary or secondary containment," from the Applicability statement for TS LCO 3.6.1.3. Delete Conditions F and G with associated Required Actions and associated completion times. Re-designate Condition H as Condition F.

E.5 TS Section 3.6.4.1, "Secondary Containment"

Delete "During CORE ALTERATIONS," and "During movement of irradiated fuel assemblies in the primary or secondary containment," from the Applicability statement for TS LCO 3.6.4.1. In addition, delete "during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or" from TS Section 3.6.4.1 Condition C. Also, delete the Note, "LCO 3.0.3 is not applicable," Required Actions C.1 and C.2, and associated completion times from Condition C. Re-designate Required Action C.3 as C.1.

E.6 TS Section 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)"

Delete "During CORE ALTERATIONS," and "During movement of irradiated fuel assemblies in the primary or secondary containment," from the Applicability statement for TS LCO 3.6.4.2. In addition, delete "during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or" from TS Section 3.6.4.2 Condition D. Also, delete the Note, "LCO 3.0.3 is not applicable," Required Actions D.1 and D.2, and associated completion times from Condition D. Re-designate Required Action D.3 as D.1.

E.7 TS Section 3.6.4.3, "Standby Gas Treatment (SGT) System"

Delete "During CORE ALTERATIONS," and "During movement of irradiated fuel assemblies in the primary or secondary containment," from the Applicability statement for TS LCO 3.6.4.3. In addition, delete "during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or" from TS Section 3.6.4.3 Conditions C and E. Also, delete the Note, "LCO 3.0.3 is not applicable," Required Actions C.2.1 and C.2.2, and associated completion times from Condition C. Re-designate Required Action C.2.3 as C.2. Additionally, delete Required Actions E.1 and E.2, as well as the associated completion times, from Condition E. Re-designate Required Action E.3 as E.1.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

Scope of Analysis

The design basis FHA was re-evaluated in order to justify the relaxation of TS requirements for the operability of certain Emergency Safety Feature (ESF) Systems while fuel is being handled and during Core Alterations. In the FHA, an irradiated fuel assembly is postulated to drop 34 feet on to the reactor core and cause the release of fission products from inside the fuel cladding that fails due to the impact. According to the CPS USAR (Reference 7), the most conservative FHA which results in the largest number of fuel rod cladding failures is in the primary containment building. The 34-foot fuel bundle drop in containment bounds the

corresponding accident in the fuel building because the fuel building FHA fuel bundle drop is only from six feet.

The analysis was based on the GE14 fuel bundle type that was conservatively assumed to have been operated at a reactor power level of 3542 Megawatts-thermal (MWt). This is 20 percent above the rated reactor power of 2894 MWt, plus an additional 2 percent power to account for uncertainty in order to provide a bounding analysis consistent with the plans to increase the reactor power level (Reference 8). The use of the GE14 fuel bundle is bounding because the other fuel bundle type in the core is GE10 fuel which has a smaller peaking factor than the GE14 fuel peaking factor. Specifically, the GE14 fuel bundle has a peaking factor of 1.7, which is much higher than the peaking factor of 1.5 for GE10. The higher peaking factor yields a higher bundle power which in turn yields a larger source term.

For the proposed TS changes, the bounding FHA was analyzed based on the AST provisions in Reference (1) for the condition when primary containment, secondary containment and SGT Systems are inoperable. The purpose of this analysis is to demonstrate that the offsite and control room dose limits as specified in Reference (3) would not be exceeded for irradiated fuel that had been allowed to decay for 24 hours. The 24-hour decay time is based on the minimum requirement for reactor subcriticality prior to fuel movement as defined in the CPS Operational Requirements Manual (ORM). By demonstrating that the dose limits will be met after a 24-hour decay period, fuel assemblies can be handled without primary and secondary containment systems operable and without the SGT System operable. No credit is taken for administrative actions to manually close doors, hatches, and penetrations in accordance with Section 11.3.6.5 of Reference (2). Even though the administrative controls are not accounted for in this analysis, they will be conservatively implemented at CPS.

Assumptions and Inputs

The key assumptions for the analysis are as follows.

1. The analysis is based on regulatory guidance in Reference (1) and Reference (9).

2. The fuel is assumed to decay for at least 24 hours.

3. The most limiting high burnup fuel type, GE14, that has a radial peaking factor of 1.7, was used to derive the source term.

4. The reactor power level is conservatively 120% of rated power plus an additional 2 percent power to account for uncertainty (i.e., 3542 MWt). This is the maximum power level anticipated from the Extended Power Uprate project. The maximum power level prior to the next refueling outage will be the current 100 % power (i.e., 2894 MWt).

5. The dual inlet CPS control room ventilation is used including the radiation monitors at the inlets consistent with the requirements of the control room ventilation system TS.

6. The existing atmospheric dispersion factors for the exclusion area boundary (EAB), the low population zone (LPZ), and the control room in the CPS USAR safety analysis continue to apply.

7. For this analysis, the limiting FHA is assumed to occur in the primary containment (Reference 7). The original, current analysis is based on a less-limiting FHA in the

secondary containment where the drop distance and associated number of failed fuel rods is less than for the FHA in the primary containment. The original analysis took credit for a very low leakage rate from primary containment. Based on this holdup of fission products in the primary containment, the FHA in the fuel building (i.e., secondary containment) resulted in higher offsite radiological releases.

8. The fuel handling accident is as described in GE's analysis (Reference 10) which results in 172 failed fuel rods that is caused by a 34-foot drop of a GE14 bundle that has a 10x10 array of fuel onto the reactor core. The CPS NF-500 fuel handling mast is assumed to fall onto the core along with the fuel bundle.

9. Although the depth of the water in containment assumed for this analysis is over 34 feet, the decontamination factors are conservatively based on a 23-foot depth of water. The TS minimum required water level above the top of the reactor pressure vessel flange is 22 ft. 8 inches.

10. The offsite dose limit is 6.3 Rem Total Effective Dose Equivalent (TEDE) (Reference 1) and the control room dose limit is 5 Rem TEDE (Reference 2) for the worst two-hour release case.

11. Primary and secondary containment isolations were not credited.

12. The SGT filtration was not credited.

13. The release of radioactive material was assumed to be directly into the wake of the containment and the ventilation flow into the control room enters through one of the two normal inlets that have radiation monitoring systems.

14. For the limiting scenario, filtration for the control room ventilation intake flow and re-circulation flow is inoperable.

15. The calculated EAB dose bounds the LPZ dose. Because the dose limits are equal, EAB dose was used for the analysis.

16. No credit is taken for atmospheric dilution or mixing in the containment.

Key inputs for the analyses of the AST FHA are in Table 1. The atmospheric dispersion factors for the control room are the NRC-approved values in the CPS USAR and are based on the release of the fission products directly into the wake of the containment building.

TABLE 1 Inputs for the AST FHA Analysis

Parameter Description	Parameter Value	Clarifying Comments
Reactor Power Level	3542 MWt	Includes 120 % extended power uprate and 2 % uncertainty
Radial Peaking Factor	1.7	For GE14 fuel. (The maximum burnup is derived from the peak licensed burnup and this peaking factor.)
Number of rods failed	172	From the drop of a fuel bundle of 34 ft. in containment
Decay period	24 hours	Based on CPS ORM
Gap Fractions: • I-131 • Kr-85 • Other Noble Gases • Other Halogens • Alkali Metals	 8% 10% 5% 5% 12%	Gap fractions from Reference 1.
Iodine Release Chemical Form: • Elemental • Organic • Aerosol	 4.85 % 0.15 % 95.0 %	Based on Reference 1
χ/Q: • EAB • Control Room	 0-2 hr 1.8E-4 s/m^3 0-2 hr 4.61E-4 s/m^3 2-8 hr 4.61E-4 s/m^3 8-24 hr 2.73E-4 s/m^3 1-4 days 9.03E-5 s/m^3 4-30 days 1.61E-5 s/m^3	The original NRC approved USAR values.
Control Room and EAB breathing rate	3.47E-4 m^3/s	Based on the CPS USAR.
Pool Water Decontamination Factor	200	Based on Reference 1
Release period	2 hr	Based on Reference 1
Control Room Recirculation Filtration Efficiency	Elemental iodine: 70% Organic iodine: 70% Particulate iodine: 70%	Based on calculation. In scenarios 3 and 4, no filtration is assumed (see Table 2).
Control Room Intake Filter Efficiency	Elemental iodine: 99% Organic iodine: 99% Particulate iodine: 99%	Based on calculation. In scenarios 3 and 4 no filtration is assumed (see Table 2).

The four scenarios in Table 2 were analyzed in order to assess the sensitivity of the dose to the control room ventilation operation and filtration levels. The control room HVAC system is designed to maintain a habitable environment under accident conditions by providing adequate protection against radiation and toxic gases. The system is provided with radiation monitors to monitor radiation levels at the minimum outside air intakes and upon detecting a radiation level exceeding a preset value automatically limits the introduction of contaminants into the system by filtering the contaminated air. This ventilation system design coupled with the control room shielding assures that the dose to the operators inside the control room is within the limits specified by Criterion 19 of 10 CFR 50 Appendix A for the duration of a design-basis accident.

TABLE 2 Scenarios for the AST Analysis

Scenario No.	Scenario Timing Description	Clarifying Comments
1	T=0 to T=20 minutes • Control Room (CR) intake flow: 0 cfm • CR unfiltered flow in-leakage: 1510 cfm • CR filtered in-leakage (via recirculation filters): 0 cfm • CR filtered recirculation flow: 0 cfm T=20 minutes to T=30 days • CR intake flow: 3300 cfm • CR unfiltered inleakage: 10 cfm • CR filtered in-leakage (via recirculation filters): 650 cfm • CR filtered recirculation flow: 54,900 cfm	With intake and recirculation filters operable
2.	T=0 hr to T= 30 days • CR filtered intake flow: 3920 cfm • CR filtered in-leakage: 10 cfm • CR filtered in-leakage: (via recirculation flow): 650 cfm • CR filtered recirculation flow: 54,900 cfm	With intake and recirculation filters operable
3.	T=0 to T=20 minutes • Control Room (CR) intake flow: 0 cfm • CR unfiltered flow in-leakage: 1510 cfm • CR filtered in-leakage (via recirculation filters): 0 cfm • CR unfiltered recirculation flow: 0 cfm T=20 minutes to T=30 days • CR unfiltered intake flow: 3300 cfm • CR unfiltered in-leakage: 10 cfm • CR unfiltered in-leakage (via recirculation filters): 650 cfm • CR unfiltered recirculation flow: 54,900 cfm	Without intake and recirculation filtration
4.	T=0 hr to T= 30 days • CR unfiltered intake flow: 4400 cfm • CR unfiltered in-leakage flow: 660 cfm	Without intake and recirculation filtration

The fission product source from the fuel at time equals zero (i.e., T=0) when the reactor was shutdown was calculated by GE using the ORIGEN2 computer code (Reference 11). The T=0 source term was adjusted for a 24 hour decay time using the ORIGEN-S (Reference 12) computer code. The dose to control room operators and the dose at the EAB was calculated using the above inputs in the RADTRAD 3.02a computer code (Reference 13).

Results

In identifying these proposed changes, AmerGen revised the design basis FHA analysis. The radiological consequences for the scenarios in Table 2, as determined by the revised analysis, are provided in Table 3. Table 3 shows the sensitivity of control room dose to equipment operation. In addition, Table 3 indicates that the EAB and control room dose meet the regulatory limits when primary containment, secondary containment, and SGT systems are not operable for the worst case FHA. The scenario with the highest operator dose was Scenario No. 4 that assumed the emergency ventilation mode was not initiated and that there was no filtration for the intake or for recirculation flow.

TABLE 3 EAB and Control Room Dose (TEDE)

Receptor	Calculated Dose (Rem)	Regulatory Limit (Rem)
EAB (0-2 hr)	0.45	6.3 (see Reference 1)
Control Room (0-30days):		
– Scenario No. 1	0.06	5
– Scenario No. 2	0.03	5
– Scenario No. 3	0.50	5
– Scenario No. 4	0.80	5 (see Reference 3)

To form the new licensing basis for CPS, the EAB dose is rounded up to 1 Rem TEDE, the LPZ dose is rounded up to 1 Rem TEDE, and the control room dose is rounded up to 1 Rem TEDE.

In conclusion, the analysis indicates that the primary containment and associated radiation monitoring systems, secondary containment and associated radiation monitoring system, the SGT System, and the intake and recirculation filtration system for the main control room ventilation system are not required in order to meet the EAB, LPZ, and control room dose regulatory limits for the design basis FHA.

G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding impact on any previous submittals, and have determined that there is an impact on the Extended Power Uprate (EPU) amendment request in Reference 8. As noted above, this analysis has been performed based on a power level consistent with that proposed in the CPS EPU submittal. However, approval of the proposed CPS Technical Specifications changes is independent of the approval of the EPU amendment request.

H. SCHEDULE REQUIREMENTS

We request approval of these proposed changes prior to January 20, 2002, with a 60-day implementation period to support preparation for implementation during the next refueling outage. The next refueling outage is currently scheduled to begin March 22, 2002.

I. REFERENCES

(1) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000.

(2) NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.

(3) Code of Federal Regulations, Title 10, Part 50.67, "Domestic Licensing of Production and Utilization Facilities, Accident Source Term."

(4) Letter from D. Hood (U. S. NRC) to G. Van Middlesworth (Nuclear Management Company, LLC), "Duane Arnold Energy Center – Issuance of Amendment Regarding Secondary Containment Operability During Movement of Irradiated Fuel and Core Alterations (TAC No. MB1569)," dated April 6, 2001.

(5) Letter from S. Sekerak (U. S. NRC) to W. Eaton (Entergy Operations), "Grand Gulf Nuclear Station (GGNS), Unit 1 – Issuance of Amendment Re: Full-Scope Implementation of an Alternative Accident Source Term (TAC No. MA8065)," dated March 14, 2001.

(6) Technical Specification Task Force (TSTF) Traveler TSTF-51, Rev. 2, "Revised Containment Requirements During Handling of Irradiated Fuel and During Core Alterations," GE-NE-A22-00110-56-01, "Boiling Water Reactor Owners Group, Technical Specification Task Force," dated October 1999.

(7) CPS Updated Safety Analysis Report, 15.7.4, "Fuel Handling Accident."

(8) Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U. S. NRC, "Request for License Amendment for Power Uprate Operation," dated June 18, 2001.

(9) NRC Standard Review Plan, 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Rev. 0.

(10) General Electric document NEDE-24011-P-A-14-US, "General Electric Standard Application for Reactor Fuel," dated June 2000.

(11) Oak Ridge National Laboratory document ORNL/TM-11018, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," S. B. Ludwig and J. P. Renier, December, 1989.

(12) NUREG/CR-200, Volume 2, Section F7, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms."

(13) NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," dated June 8, 1999.

Attachment B
Proposed Technical Specification Changes
Clinton Power Station, Unit 1

MARKED-UP TS PAGES FOR THE PROPOSED CHANGES

MARKED-UP AND REVISED TS PAGES

3.3-53	3.3-57	3.3-59
3.3-64	3.6-3	3.6-7
3.6-9	3.6-14	3.6-15
3.6-43	3.6-44	3.6-47
3.6-49	3.6-51	3.6-53

MARKED-UP BASES PAGES
(PROVIDED FOR INFORMATION ONLY)

B 3.3-146a	B 3.3-148	B 3.3-157
B 3.3-158	B 3.3-169	B 3.3-179
B 3.3-180	B 3.6-8	B 3.6-12
B 3.6-13	B 3.6-22	B 3.6-22a
B 3.6-25	B 3.6-85	B 3.6-86
B 3.6-87	B 3.6-90	B 3.6-91
B 3.6-93	B 3.6-98	B 3.6-99
		B 3.6-100

3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Locks

LCO 3.6.1.2 Each primary containment air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

----------NOTE----------
The following Applicabilities apply only to the upper containment personnel air lock.

During CORE ALTERATIONS,
During movement of irradiated fuel assemblies in the primary or secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------
1. Entry and exit is permissible to perform repairs of the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Locks

LCO 3.6.1.2 Each primary containment air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

----------NOTE----------
The following Applicabilities apply only to the upper containment personnel air lock.

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------
1. Entry and exit is permissible to perform repairs of the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. As required by Required Action F.1 and referenced in Table 3.3.6.1-1.	N.1 Isolate the affected penetration flow path(s).	Immediately
	OR	
	N.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	N.2.2 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
N.2	N.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
O. As required by Required Action F.1 and referenced in Table 3.3.6.1-1.	O.1 Isolate the affected penetration flow path(s).	Immediately
	OR	
	O.2 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. As required by Required Action F.1 and referenced in Table 3.3.6.1-1.	N.1 Isolate the affected penetration flow path(s).	Immediately
	OR	
	N.2 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
O. As required by Required Action F.1 and referenced in Table 3.3.6.1-1.	O.1 Isolate the affected penetration flow path(s).	Immediately
	OR	
	O.2 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation (continued)					
g. Containment Building Fuel Transfer Pool Ventilation Plenum Radiation-High	(c), (d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 500 mR/hr
h. Containment Building Exhaust Radiation-High	1,2,3	4[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c), (d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
i. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High	1,2,3	4[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c), (d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
j. Reactor Vessel Water Level-Low Low Low, Level 1	1,2,3	4[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
	(c) d	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
k. Containment Pressure-High	(e)	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3.0 psid
l. Manual Initiation	1,2,3	2[b]	J	SR 3.3.6.1.6	NA
	(c), (d)	2	N	SR 3.3.6.1.6	NA

(continued)

(b) Also required to initiate the associated drywell isolation function.

(c) During operations with a potential for draining the reactor vessel.

(d) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment.

(e) MODES 1, 2, and 3 with the associated PCIVs not closed.

d

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation (continued)					
g. Containment Building Fuel Transfer Pool Ventilation Plenum Radiation-High	(c)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 500 mR/hr
h. Containment Building Exhaust Radiation-High	1,2,3	4[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
i. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High	1,2,3	4[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
j. Reactor Vessel Water Level-Low Low Low, Level 1	1,2,3	4[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
k. Containment Pressure-High	(d)	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3.0 psid
l. Manual Initiation	1,2,3	2[b]	J	SR 3.3.6.1.6	NA
	(c)	2	N	SR 3.3.6.1.6	NA

(continued)

(b) Also required to initiate the associated drywell isolation function.

(c) During operations with a potential for draining the reactor vessel.

(d) MODES 1, 2, and 3 with the associated PCIVs not closed.

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
j. Drywell Pressure - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
k. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 66.1 gpm
b. Differential Flow-Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 47 seconds
c. RWCU Heat Exchanger Equipment Room Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 205°F
d. RWCU Pump Rooms Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 202°F
e. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
g. Standby Liquid Control System Initiation	1,2	2	L	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
	(c) (d)	2	N	SR 3.3.6.1.6	NA

(continued)

(c) During operations with a potential for draining the reactor vessel.

(d) During CORE ALTERATIONS and during ovement of irradiated fuel assemblies in the primary or secondary containment.

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
j. Drywell Pressure - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
k. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 66.1 gpm
b. Differential Flow-Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 47 seconds
c. RWCU Heat Exchanger Equipment Room Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 205°F
d. RWCU Pump Rooms Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 202°F
e. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
g. Standby Liquid Control System Initiation	1,2	2	L	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
	(c)	2	N	SR 3.3.6.1.6	NA

(continued)

(c) During operations with a potential for draining the reactor vessel.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Reactor Vessel Water Level-Low Low, Level 2	1,2,3,(a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ -47.7 inches
2.	Drywell Pressure-High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.88 psig
3.	Containment Building Fuel Transfer Pool Ventilation Plenum Exhaust Radiation-High	(a), (b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 500 mR/hr
4.	Containment Building Exhaust Radiation-High	~~1,2,3,~~ (a), (b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 400 mR/hr
5.	Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High	~~1,2,3,~~ (a), (b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 400 mR/hr
6.	Fuel Building Exhaust Radiation-High	~~1,2,3,~~ (c)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 17 mR/hr
7.	Manual Initiation	~~1,2,3,~~ (a), (b)	1	SR 3.3.6.2.5	NA

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment.

(c) During movement of irradiated fuel assemblies in the fuel building.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Reactor Vessel Water Level-Low Low, Level 2	1,2,3,(a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ -47.7 inches
2.	Drywell Pressure-High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.88 psig
3.	Containment Building Fuel Transfer Pool Ventilation Plenum Exhaust Radiation-High	(a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 500 mR/hr
4.	Containment Building Exhaust Radiation-High	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 400 mR/hr
5.	Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 400 mR/hr
6.	Fuel Building Exhaust Radiation-High	1,2,3,	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 17 mR/hr
7.	Manual Initiation	1,2,3, (a)	1	SR 3.3.6.2.5	NA

(a) During operations with a potential for draining the reactor vessel.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, or C not met during movement of irradiated fuel assemblies in the primary or secondary containment, CORE ALTERATIONS, or OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	E.1 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiation action to suspend OPDRVs.	Immediately

E.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, or C not met during OPDRVs.	E.1 Initiation action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 for RHR Shutdown Cooling System suction from the reactor vessel isolation valves when associated isolation instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Function 5.c.

----------NOTE----------
The following Applicabilities apply only to secondary containment bypass leakage isolation valves.

During CORE ALTERATIONS,
During movement of irradiated fuel assemblies in the primary or secondary containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------
1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 for RHR Shutdown Cooling System suction from the reactor vessel isolation valves when associated isolation instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Function 5.c.

----------*NOTE*----------
The following Applicabilities apply only to secondary containment bypass leakage isolation valves.

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------
1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.3 Perform SR 3.6.1.3.5 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
F. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during movement of irradiated fuel assemblies in the primary or secondary containment.	F.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in primary and secondary containment.	Immediately
G. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during CORE ALTERATIONS.	G.1 Suspend CORE ALTERATIONS.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.3 Perform SR 3.6.1.3.5 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, or 3.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F → H. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5 or during OPDRVs.	H.1 F.1 Initiate action to suspend OPDRVs.	Immediately
	OR	
	H.2 F.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	----------NOTES---------- 1. Only required to be met in MODES 1, 2, and 3. 2. Not required to be met when the 36 inch primary containment purge valves are open for pressure control, ALARA or air quality considerations for personnel entry. Also not required to be met during Surveillances or special testing on the purge system that requires the valves to be open. The 36 inch primary containment purge lines shall not be opened with a 12 inch primary containment purge line open nor with a drywell vent and purge supply or exhaust line open. ---------- Verify each 36 inch primary containment purge valve is closed.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, or D not met for PCIV(s) required to be OPERABLE during MODE 4 or 5 or during OPDRVs.	F.1 Initiate action to suspend OPDRVs. OR F.2 Initiate action to restore valve(s) to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	----------NOTES---------- 1. Only required to be met in MODES 1, 2, and 3. 2. Not required to be met when the 36 inch primary containment purge valves are open for pressure control, ALARA or air quality considerations for personnel entry. Also not required to be met during Surveillances or special testing on the purge system that requires the valves to be open. The 36 inch primary containment purge lines shall not be opened with a 12 inch primary containment purge line open nor with a drywell vent and purge supply or exhaust line open. ---------- Verify each 36 inch primary containment purge valve is closed.	31 days

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the primary or secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.3 Initiate action to suspend OPDRVs.	Immediately

C.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable during OPDRVs.	C.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the primary or secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

----------NOTES----------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	D.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D.3 Initiate action to suspend OPDRVs.	Immediately

D.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during OPDRVs.	D.1 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the primary or secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	C.1 Place OPERABLE SGT subsystem in operation.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
C.2	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during OPDRVs.	C.1 Place OPERABLE SGT subsystem in operation. OR C.2 Initiate action to suspend OPDRVs.	Immediately Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1 Suspend movement of irradiated fuel assemblies in the primary and secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

E.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	18 months
SR 3.6.4.3.4	Verify each SGT filter cooling bypass damper can be opened and the fan started.	18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two SGT subsystems inoperable during OPDRVs.	E.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	18 months
SR 3.6.4.3.4	Verify each SGT filter cooling bypass damper can be opened and the fan started.	18 months

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.g., 2.h and 2.i. Containment Building Fuel Transfer Pool Ventilation Plenum, Containment Building, and Containment Building Continuous Containment Purge (CCP) Exhaust Radiation-High (continued)

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.

These Functions are required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the primary or secondary 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.

2.j. Reactor Vessel Water Level-Low Low Low, Level 1

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the primary containment occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level-Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level-Low Low Low, Level 1 Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA. In addition, this Function provides an isolation signal to certain drywell isolation valves. The isolation of drywell isolation

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.k. Containment Pressure-High (continued)

The Allowable Value was chosen to prevent opening of the containment ventilation supply and exhaust isolation bypass valves when excessive differential pressure could result in damage to the associated ductwork.

Two channels of the Containment Pressure-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

2.l. 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 two push buttons for the logic, one manual initiation push button per trip system (i.e., 1B21H-S25A and 1B21H-S25B). There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of the Manual Initiation Function are available and are required to be OPERABLE. This Function is also required to be OPERABLE during CORE ALTERATIONS, movement of irradiated fuel assemblies in primary or secondary containment, or OPDRVs. This Function initiates isolation of valves which isolate primary containment penetrations which bypass secondary containment. Thus, this Function is also required under those conditions in which secondary containment is required to be OPERABLE.

3. Reactor Core Isolation Cooling System Isolation

3.a. Auxiliary Building RCIC Steam Line Flow-High

Auxiliary Building 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,

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.f. Reactor Vessel Water Level-Low Low, Level 2 (continued)

containment. Thus, this Function is also required under those conditions in which a low reactor water level signal could be generated when secondary containment is required to be OPERABLE.

4.g. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the two SLC pump start signals.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels (one from each pump) of SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

4.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the RWCU System 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 plant licensing basis.

There are two push buttons for the logic, one manual initiation push button 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.

Two channels of the Manual Initiation Function are available and are required to be OPERABLE. This Function is also required to be OPERABLE during CORE ALTERATIONS, movement of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.h. Manual Initiation (continued)

irradiated fuel assemblies in primary or secondary containment, or OPDRVs. This Function initiates isolation of valves which isolate primary containment penetrations which bypass secondary containment. Thus, this Function is also required under those conditions in which secondary containment is required to be operable.

5. RHR System Isolation

5.a. Ambient Temperature-High

Ambient Temperature-High is provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. This Function is not assumed in any USAR transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

Ambient Temperature-High signals are initiated from thermocouples that are appropriately located to protect the system that is being monitored. Two instruments monitor each area. Four channels for RHR Ambient Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

The RHR Equipment Room Ambient Temperature-High Function is only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, insufficient pressure and temperature are available to develop a significant steam ~~leak in this piping and~~ significant water leakage is protected by the Reactor Vessel Water Level-Low, Level 3 Function.

5.b, 5.c. Reactor Vessel Water Level-Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor or vessel interfaces occurs to begin isolating the

(continued)

BASES

ACTIONS

M.1, M.2, M.3.1, M.3.2, M.3.3, and M.3.4 (continued)

radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillances may need to be performed to restore the component to OPERABLE status. In addition, at least one door in the upper containment personnel air lock must be closed. The closed air lock door completes the boundary for control of potential radioactive releases. With the appropriate administrative controls however, the closed door can be opened intermittently for entry and exit. This allowance is acceptable due to the need for containment access and due to the slow progression of events which may result from a reactor vessel draindown event. Reactor vessel draindown events would not be expected to result in the immediate release of appreciable fission products to the containment atmosphere. Actions must continue until all requirements of this Condition are satisfied.

N.1, N.2.1, N.2.2, N.2.3, O.1, and O.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 N.1 or O.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, CORE ALTERATIONS and movement of irradiated fuel assemblies 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 OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission production release. Actions must continue until OPDRVs are suspended.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4, 5, 6. Containment Building Fuel Transfer Pool Ventilation Plenum, Containment Building, Containment Building Continuous Containment Purge (CCP), and Fuel Building Exhaust Radiation-High (continued)

detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of each of these Exhaust Radiation-High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Exhaust Radiation-High High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the primary containment or fuel building, as applicable, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery ~~or dropped fuel assemblies~~) must be provided to ensure that offsite dose limits are not exceeded.

7. Manual Initiation

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

There are two push buttons for the logic, one manual initiation push button 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)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Manual Initiation (continued)

Two channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3 and during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the primary containment or Fuel Building, since these are the MODES and other specified conditions in which the Secondary Containment Isolation automatic Functions are required to be OPERABLE.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation 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 secondary containment isolation 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 secondary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 24 hours has been shown to be acceptable (Refs. 3 and 4) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the

(continued)

BASES

LCO (continued)

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 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 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), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the primary containment.

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. 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 performed from the barrel side of ~~the door, then it is~~ permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the primary containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

(continued)

BASES

ACTIONS

C.1, C.2 and C.3 (continued)

both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected primary containment air locks must be verified closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in each affected air lock.

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, E.2, and E.3

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time during OPDRVs, during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the primary or secondary containment, action is required to immediately suspend activities that represent a potential for releasing radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

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. Action must continue until OPDRVs are suspended.

The Required Actions of Condition E are 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 sufficient reason to require a reactor shutdown.

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 during 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 overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

The SR has been modified by three Notes. Note 1 provides an exception to the specific leakage requirements for the primary containment air locks in other than MODES 1, 2, and 3. When not operating in MODES 1, 2, or 3, primary containment pressure is not expected to significantly increase above normal, and therefore specific testing at elevated pressure is not required. Note 2 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 3 has been added to this SR, requiring the results to be evaluated against the acceptance criteria applicable to SR 3.6.1.1.1, i.e., the acceptance criteria

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

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.

F.1 and F.2

F.1, G.1, H.1, and H.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, CORE ALTERATIONS and movement of irradiated fuel assemblies in the primary and secondary 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 valve.

IF

The Required Actions of Condition F are 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 sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

This SR verifies that the 36-inch primary containment purge 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 leakage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

outside of the limits. If the open valve is known to have excessive leakage, Condition D applies.

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. If a LOCA inside primary containment occurs in these MODES, the primary containment purge valves are capable of closing before the pressure pulse affects systems downstream of the purge valves and the release of radioactive material will not exceed limits prior to the purge valves closing. 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 irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.4 (continued)

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.

With regard to isolation time values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 8).

SR 3.6.1.3.5

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of the Primary Containment Leakage Rate Testing Program is required to ensure OPERABILITY. The acceptance criterion for this test is ≤ 0.01 L_a when pressurized to Pa, 9.0 psig. Since cycling these valves may introduce additional seal degradation (beyond that which occurs to a valve that has not been opened), this SR must be performed within 92 days after opening the valve. However, operating experience has demonstrated that if a valve with a resilient seal is not stroked during an operating cycle, significant increased leakage through the valve is not observed. Based on this observation, a normal Frequency in accordance with the Primary Containment Leakage Rate Testing Program was established.

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 irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

With regard to leakage rate values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 9).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Secondary containment satisfies Criterion 3 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.

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 operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the primary or secondary containment.

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

(continued)

BASES

ACTIONS

A.1 (continued)

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.

C.1, C.2, and C.3

Movement of irradiated fuel assemblies in the primary or secondary containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. CORE ALTERATIONS and movement of irradiated fuel assemblies 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, 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.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend

(continued)

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

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.

With regard to secondary containment vacuum values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 3).

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that secondary containment equipment hatches and 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. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the boundary established by SCIDs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIDs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIDs satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

The power operated isolation dampers and valves are considered OPERABLE when their isolation times are within limits. Additionally, power operated automatic dampers and valves are required to actuate on an automatic isolation signal.

The normally closed isolation dampers, valves, or blind flanges are considered OPERABLE when manual dampers or valves are closed or open in accordance with appropriate administrative controls, automatic dampers are de-activated and secured in their closed position, or blind flanges are in place. The SCIDs covered by this LCO, along with their associated stroke times, if applicable, are listed in applicable plant procedures.

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 SCIDs 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 SCIDs 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 operations with a potential for draining the reactor vessel (OPDRVs), during

(continued)

BASES

APPLICABILITY (continued)

CORE ALTERATIONS, or during movement of irradiated fuel assemblies. Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

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 individual, 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 SCID. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIDs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCID.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCID inoperable, the affected penetration flow path(s) must be isolated. 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 criteria are a closed and de-activated automatic damper, a closed manual damper or valve, or a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. This Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIDs to close, occurring during this short time.

(continued)

ACTIONS

C.1 and C.2 (continued)

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, D.2, and D.3

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, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the primary and secondary 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.

If

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies each secondary containment isolation manual valve, damper, 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 valve or damper manipulation. Rather, it involves verification that those SCIDs in secondary containment that are capable of being mispositioned are in the correct position.

(continued)

BASES

APPLICABILITY (continued)

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 SGT System 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 operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the primary or secondary containment.

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE SGT 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 SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT 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, C.2.2, and C.2.3 and C.2

During movement of irradiated fuel assemblies in the primary or secondary containment, during CORE ALTERATIONS, or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE,

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

that no 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 radioactive material to the secondary containment, thus placing the unit in a Condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies 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 to minimize the probability of a vessel draindown and subsequent potential for fission product release. This action should be chosen if the OPDRVs could be impacted by a loss of offsite power. Action must continue until OPDRVs are suspended.

If

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

D.1

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

E.1, E.2, and E.3

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the primary and secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

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. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem from the main 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 (automatic heater cycling to maintain temperature) 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.

With regard to operating time values obtained pursuant to this SR, as read from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 9).

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber bypass leakage and efficiency, minimum system flow rate, combined HEPA filter and charcoal adsorber pressure drop, and heater dissipation. The frequencies for performing the SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4) and include testing initially, after 720 hours of system operation, once per 18 months, and following painting, fire, or chemical release in any ventilation zone communicating with the system. The laboratory test results will be

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INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92, "Issuance of Amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated; or,

(2) Create the possibility of a new or different kind of accident from any previously analyzed; or,

(3) Involve a significant reduction in a margin of safety.

AmerGen Energy Company, LLC (i.e., AmerGen), proposes changes to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS), Unit 1. Specifically, we propose to remove the application of "during Core Alterations" and "during handling of irradiated fuel in primary and secondary containment" from the following TS sections, as applicable.

- 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation
- 3.3.6.2, Secondary Containment Isolation Instrumentation
- 3.6.1.2, Primary Containment Air Locks
- 3.6.1.3, Primary Containment Isolation Valves (PCIV)
- 3.6.4.1, Secondary Containment
- 3.6.4.2, Secondary Containment Isolation Dampers (SCIDs)
- 3.6.4.3, Standby Gas Treatment (SGT) System

These proposed changes are based on a revised analysis of the Fuel Handling Accident (FHA) for CPS that conforms to the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and provides a new design and licensing basis for the FHA based on the Alternative Source Term (AST).

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for these proposed changes is indicated below.

Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The equipment affected by the proposed changes are mitigative in nature, and relied upon after an accident has been initiated. Application of the Alternative Source Term (AST) does not involve a change to the plant design. While the operation of the primary and secondary containment systems do change as a result of these proposed changes, these systems are not accident initiators. Application of the AST does not initiate a design basis accident. Similarly, application of the AST does not affect the design or operation for any equipment or systems involved in the mitigation of accidents. The proposed changes to the Technical Specifications (TS), while they revise certain performance requirements, do not involve any physical modifications to the plant. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed.

The AST changes do not affect the design and operation of the facility. Rather, once the accident has been postulated the new source term is an input to the evaluation of the consequences. The implementation of the AST has been evaluated in revisions to the analyses of the worst case Fuel Handling Accident (FHA) at Clinton Power Station (CPS). Based on the results of the analyses, it has been demonstrated that, with the proposed changes, the dose consequences of the worst case FHA remain a small fraction of the regulatory guidance provided by the NRC for the AST in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000. Since the primary containment systems, secondary containment systems and the Standby Gas Treatment (SGT) system are not assumed to be operable in the FHA, the consequences of eliminating the requirements that these systems be operable during the handling of irradiated fuel in both primary and secondary containment or during core alterations will not increase significantly.

In summary, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No new equipment is introduced, and no installed equipment is operated in a new or different manner. There is no change to the predicted accident response of any plant structure, system or component. The proposed change in availability of mitigative equipment has been evaluated in accordance with the guidance in RG 1.183 and does not produce different or more limiting accident progression or results. As such, no new accident modes or equipment failure modes are created by these proposed changes.

Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the change involve a significant reduction in a margin of safety?

The proposed changes involve a selective application of the AST for the FHA consistent with the guidance provided in RG 1.183. The existing analyses demonstrated that the dose consequences associated with the FHA were within the applicable NRC specified limits. For offsite dose, the margin to safety for the FHA using the 10 CFR 100, "Reactor Site Criteria," limits was maintained by the existing analysis. For the Control Room dose, the margin of safety using the 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion 19, "Control room," dose limits was conservatively maintained by the existing analyses. The results of the FHA analysis revised in support of this submittal however, are subject to revised acceptance criteria. The revised dose consequences of the limiting design basis FHA are within the acceptance criteria found in RG 1.183 and 10 CFR 50.67, "Domestic Licensing of Production and Utilization Facilities, Accident Source Term." The proposed changes ensure that the doses at the exclusion area boundary (EAB), low population zone (LPZ), and control room remain a small fraction of the new regulatory limits in RG 1.183 and 10 CFR 50.67.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above evaluation, we have concluded that the proposed changes do not involve a significant hazards consideration.

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

AmerGen Energy Company, LLC (i.e., AmerGen) has evaluated these proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." AmerGen has determined that these proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that these changes are being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or surveillance requirement, and the amendment meets the following specific criteria.

(i) The proposed changes involve no significant hazards consideration.

As demonstrated in Attachment C, these proposed changes do not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed changes, which remove the requirement for primary containment systems, secondary containment systems, and the Standby Gas Treatment (SGT) system during core alterations and movement of irradiated fuel in the primary and secondary containments are consistent with the design basis of the plant. There will be no increase in the amounts of any effluents released offsite. These changes do not result in an increase in power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not affect the types or increase the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the configuration of the facility. The proposed changes only affect operation of the plant in that they provide for removal of primary containment, secondary containment, and SGT system restriction while moving fuel. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from these changes.