Union Electric
Callaway Plant

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DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
PROPOSED REVISION TO TECHNICAL SPECIFICATIONS 3.9.4 "CONTAINMENT PENETRATIONS"; 3.3.6 "CONTAINMENT PURGE ISOLATION INSTRUMENTATION" AND 3.3.7 "CREVS ACTUATION INSTRUMENTATION"
TO ALLOW OPEN CONTAINMENT EQUIPMENT HATCH
DURING REFUELING OPERATIONS

Pursuant to 10 CFR 50.90, AmerenUE hereby requests an amendment to the Facility Operating License No. NPF-30 for Callaway Plant. The amendment request incorporates the attached changes into the Callaway Plant Technical Specifications.

This amendment application would revise Technical Specifications (TS) 3.9.4, "Containment Penetrations," to allow the containment equipment hatch and the emergency airlock to be open during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment under administrative controls.

TS 3.3.6, "Containment Purge Isolation Instrumentation" is revised to bypass the requirement for an automatic actuation of containment purge isolation (CPIS) during CORE ALTERATIONS and/or during movement of irradiated fuel within containment. As designed, the purge system radiation monitor detectors GTRE0022 and GTRE0033 provide automatic CPIS actuation on high radiation in containment. In the event of a fuel handling accident, during CORE ALTERATIONS and/or during the movement of irradiated fuel assemblies in containment and with an open containment equipment hatch, the containment purge system would not isolate on an automatic CPIS and would remain in operation, ensuring negative pressure in containment. In this postulated plant condition, the operable shutdown purge system also ensures that any radioactive release paths are

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directed and monitored via the shutdown purge exhaust system, until unisolated containment penetrations are closed.

TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation" is revised to add a new surveillance requirement to response time test the control room radiation monitor detector channels. By design at Callaway Plant, bypassing the automatic CPIS actuation during CORE ALTERATIONS and/or during movement of irradiated fuel within containment also eliminates the automatic CRVIS actuation based on the purge system radiation monitor detectors. However, control room radiation monitor detectors, GKRE0004 and GKRE0005, provide the primary means to ensure that CRVIS actuation occurs on high radiation following a postulated fuel handling accident.

The appropriate TS Bases changes for all proposed specification revisions are included for information and reflect the proposed changes.

Attachment 1 to this submittal provides the required Affidavit. Attachment 2 provides a detailed description, safety analysis of the proposed changes, and the Callaway determination that the proposed change does not involve a significant hazard consideration. Attachment 3 provides the existing TS pages marked-up to show the proposed change. Attachment 4 provides a clean copy of the proposed Technical Specification pages. Attachment 5 provides the existing TS Bases pages marked-up to show the proposed changes (for information only). Finally, Attachment 6 provides FSAR revisions to incorporate the proposed changes (for information only).

This letter identifies actions committed to by AmerenUE and Callaway Plant in this submittal. Other statements are provided for information purposes and are not considered to be commitments. A summary of the regulatory commitments included in this submittal is provided in Attachment 7.

The Callaway Plant Review Committee and the Nuclear Safety Review Board have reviewed and approved this amendment application. It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. In addition, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of this amendment.

AmerenUE requests approval of this proposed License Amendment by August 2002 prior to the next scheduled Refueling Outage 12. Receipt of this Amendment is not required to conduct the outage or to restart the unit following the outage. However, implementation of the requested TS change prior to the outage will allow critical planned outage work to proceed in conjunction with fuel handling activities. The approved amendment will be implemented prior to entry into MODE 6 during Refueling Outage 12.

AmerenUE is submitting this License Amendment Request in conjunction with the industry consortium of plants as a result of a mutual agreement known as Strategic Teaming and Resource Sharing or STARS. The STARS group consists of plants operated by TXU Electric, AmerenUE, Wolf Creek Nuclear Operating Corporation, Pacific Gas and Electric and STP Nuclear Operating Company. Members of the STARS group have already submitted plant specific License Amendment Requests similar to this request. The AmerenUE submittal has been coordinated with the other STARS submittals, however, takes a slightly different approach specific to Callaway Plant. The Callaway submittal is consistent with other STARS submittals in that the open containment equipment hatch provides outage flexibility without changing any dose consequences of design basis accidents. However, the method of managing and monitoring any potential release, in the event of a fuel handling accident, is specific to Callaway Plant.

Pursuant to 10 CFR 50.91(b)(1), AmerenUE is providing the State of Missouri with a copy of this proposed amendment.

If you should have any questions on the above or attached, please contact Dave Shafer at (314) 554-3104 or Dwyla Walker at (314) 554-2126.

Very truly yours,

JBlosser

John D. Blosser
Manager, Regulatory Affairs

DJW/jdg

Attachments:
1) Affidavit
2) Evaluation
3) Markup of Technical Specification pages
4) Retyped Technical Specification pages
5) Markup of Technical Specification Bases pages (for information only)
6) Markup of Callaway FSAR pages (for information only)
7) Summary of Regulatory Commitments

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Revised 11/7/01

STATE OF MISSOURI)
) SS
CALLAWAY COUNTY)

John D. Blosser, of lawful age, being first duly sworn upon oath says that he is Manager Regulatory Affairs, for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Blosser
John D. Blosser
Manager Regulatory Affairs

SUBSCRIBED and sworn to before me this 6th day

of December, 2001.

Gloria J Taylor

GLORIA J. TAYLOR
NOTARY PUBLIC
STATE OF MISSOURI - CALLAWAY COUNTY
NOTARY SEAL
MY COMMISSION EXPIRES JUNE 21, 2003

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ATTACHMENT 2

EVALUATION

EVALUATION

1.0 INTRODUCTION

This letter is a request to amend Operating License NPF-30 for Callaway Plant.

This amendment application would revise Technical Specifications (TS) 3.9.4, "Containment Penetrations," to allow the containment equipment hatch and the emergency airlock to be open during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment under administrative controls.

In addition TS 3.3.6, "Containment Purge Isolation Instrumentation," would be revised to bypass a requirement for automatic actuation of containment purge isolation (CPIS) during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment. Eliminating the automatic actuation allows the containment purge system to remain in operation during refueling when the equipment hatch is open. In the event of a postulated design basis accident, while the containment equipment hatch and/or any other containment penetrations are unisolated, an operational containment purge and exhaust system ensures that any potential radioactive release would be directed and monitored via the purge exhaust and not released directly to the environment. It also ensures negative pressure in containment is maintained. When the containment purge system remains in operation, it ensures that Callaway meets General Design Criteria 64 (GDC 64) which requires monitoring the reactor containment atmosphere and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Manual capability for isolation of the containment purge system, if required, would remain unchanged.

TS 3.3.7, "CREVS Actuation Instrumentation," is revised to add a new surveillance to response time test the channels for the control room radiation monitor detectors. During CORE ALTERATIONS and/or movement of irradiated fuel assemblies within containment, and with an open equipment hatch, the control room radiation monitor detectors provide the primary means for control room ventilation isolation (CRVIS). The surveillance verifies that the control room radiation monitor detectors function within required time limits.

TS 3.3.7 LCO and APPLICABILITY are also revised to incorporate changes into Table 3.3.7-1. The revised Table 3.3.7-1 reflects separate instrumentation requirements for Function, Surveillance Requirements and Nominal Trip Setpoint to ensure CRVIS actuation when moving irradiated fuel assemblies within the fuel building versus when moving irradiated fuel assemblies within containment. Separate actuation instrumentation is credited for protecting the control room based on generating CRVIS depending on whether a postulated fuel handling accident occurs in the fuel building or within containment.

Appropriate TS Bases and FSAR changes are included to reflect the proposed changes.

The proposed changes will allow scheduling of many outage activities to be accomplished when the reactor vessel is open and covered by 23 feet of water. The proposed changes will permit flexibility for outage work to proceed in conjunction with critical path activities without a reduction in plant safety or safety to the general public.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change would revise Limiting Condition for Operation (LCO) 3.9.4 to allow the containment equipment hatch and the emergency airlock to be open during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment, provided that they are capable of being closed. A new Surveillance Requirement (SR) would be added to verify the capability to install the containment equipment hatch, if the hatch is open, at a Frequency of seven days.

The TS Bases are revised to reflect the changes to LCO 3.9.4 and the addition of a new Surveillance Requirement. TS 3.9.4 currently includes a requirement (SR 3.9.4.2) for verifying that each required containment purge isolation valve actuates to the isolation position on an actual or simulated actuation signal. Because the automatic actuation requirement for containment purge isolation valves is bypassed, Surveillance Requirement 3.9.4.2 (renumbered to SR 3.9.4.3 after the addition of the new Surveillance) is modified to eliminate the requirement to verify an actual or simulated actuation signal. The revised requirement verifies a manual actuation signal. During CORE ALTERATIONS and/or during movement of irradiated assemblies within containment, containment purge isolation valves are OPERABLE if capable of being closed on a manual actuation signal. Finally, the Bases are revised to identify the administrative controls associated with the allowance to maintain the equipment hatch open.

Another proposed change to LCO 3.9.4 would allow the emergency air lock to remain open during refueling under administrative controls similar to the personnel air lock. During a refueling outage, other work inside containment continues during fuel movement and CORE ALTERATIONS. A large number of personnel are working in containment. An open emergency air lock adds flexibility to the scheduling of work, the movement of equipment and aids the more efficient and rapid evacuation of containment, if required. In addition, in the event that normal power is not available, the open emergency hatch provides a pathway for the cables associated with the diesel generator backup power supply that is needed to close the containment equipment hatch. Although it is very unlikely that loss of power would occur during a postulated fuel handling accident and although this scenario is beyond licensing basis, Callaway considers it prudent to provide a means for closure of the containment equipment hatch. Because under the proposed change, containment could be isolated and evacuated more quickly the dose to workers, in the event of an accident, would be reduced while the potential dose to the public would remain acceptable.

The proposed change to TS 3.3.6 revises the Limiting Condition for Operation (LCO) 3.3.6 (Table 3.3.6-1) to eliminate a requirement for automatic CPIS actuation based on detection of high radiation in containment by the containment purge exhaust radiation-gaseous monitors, GTRE0022 and GTRE0033. This change would be effective only during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment. Elimination of the automatic CPIS provides assurance that the containment purge system will not isolate, but remain in operation in the event of a postulated design basis fuel handling accident. In the event the equipment hatch is open, any release would be monitored via the containment purge exhaust system and containment would be maintained at negative pressure until the equipment hatch could be closed. Manual containment purge isolation capability remains as a TS requirement to ensure isolation capability when required. The alarm and indication functions associated with the radiation monitors GTRE0022 and GTRE0033 remain available.

As a result of the change to the LCO and since the manual containment purge isolation capability is retained, Condition C is modified to require action only if one or more manual channels are inoperable. The TS Bases are revised to reflect the changes to LCO 3.3.6 (Table 3.3.6-1) and Condition C.

Finally, a proposed change to the Limiting Condition for Operation (LCO) 3.3.7 (Table 3.3.7-1) incorporates a new surveillance requirement to assure CRVIS actuation on high radiation during CORE ALTERATIONS and/or during movement of irradiated fuel within containment. The new SR ensures that in the event of a design basis fuel handling accident in containment, the control room ventilation radiation monitor detectors, GKRE0004 and GKRE0005, detect high radiation and initiate CRVIS. The new SR requires response time testing on the control room ventilation radiation monitor instrument channels (a Note is added, however, that excludes response time testing of the monitor detectors themselves due to the nature of the detectors). In the event that a fuel handling accident occurs in the fuel building, the fuel building ventilation radiation monitor detectors, GGRE0027 and GGRE0028, detect high radiation and initiate CRVIS. Due to the remote location of these Fuel Building radiation monitors relative to the Control Room intake louvers, the FBVIS will isolate the Control Room prior to the post-accident radioactive plume reaching the Control Room intake louvers and the CRVIS function is not response time tested for these monitors.

The TS Bases are revised to reflect the changes to LCO 3.3.7 and the addition of the new Surveillance Requirement during CORE ALTERATIONS and/or during movement of irradiated fuel within containment.

3.0 BACKGROUND

The equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. Technical Specification 3.9.4, "Containment Penetrations," requires that the equipment hatch be closed and held in place by four bolts during fuel movement and CORE ALTERATIONS. This requirement ensures that a release of fission products within the containment will be restricted from escaping to the environment.

As described in Section 3.8.2.1.1 of the Callaway Final Safety Analysis Report (FSAR), the equipment hatch is a welded steel assembly with a double-gasketed, flanged, and bolted cover. A moveable missile shield is provided on the outside of the reactor building to protect the equipment hatch. During shutdown conditions, administrative controls ensure that an appropriate missile barrier is in place during the threat of severe weather that could result in the generation of tornado driven missiles. The equipment hatch is raised and lowered with two dedicated hoists. Each hoist is electrically powered from the normal non-class 1E electrical distribution system. Both hoists are needed to close the containment equipment hatch.

Also described in Section 3.8.2.1.1 are the personnel and emergency air locks that form part of the containment pressure boundary and provide personnel access during all MODES of operation. The personnel air lock is nominally a right circular cylinder, approximately 10-ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft 9 in inside diameter with a 2 ft 6 in door at each end. Callaway License Amendment 114 revised TS 3.9.4 and its associated Bases to allow the containment personnel airlock doors to be open during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The regulatory basis for TS 3.9.4, "Containment Penetrations," is to ensure that the primary containment is capable of containing fission product radioactivity that may be released from the reactor core following a fuel handling accident inside containment. This ensures that offsite radiation exposures are maintained well within the requirements of 10 CFR 100.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 16, "Containment Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

GDC 56, "Primary Containment Isolation," describes the isolation provisions that must be provided for lines that connect directly to the containment atmosphere and which penetrate primary reactor containment unless it can be demonstrated that the isolation provisions for a specific class of lines are acceptable on some other defined basis.

GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

GDC 64, "Monitoring Radioactivity Releases," requires monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

U. S. NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," is NRC guidance which describes a method acceptable to the NRC staff for licensee evaluation of the potential radiological consequences of a fuel handling accident.

NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," relates to the expected release fraction for the radioactive iodine. According to this report, the calculated release fraction for extended burnup fuel may be up to 20% higher than that assumed in Regulatory Guide 1.25 for iodine-131.

NUREG-0800, "U. S. NRC Standard Review Plan," Section 15.7.4, provides guidance to the NRC staff for the review and evaluation of system design features and plant procedures provided for the mitigation of the radiological consequences of postulated fuel handling accidents.

The parameters of concern and the acceptance criteria applied are based on the requirements of 10 CFR 100 with respect to the calculated radiological consequences of a fuel handling accident and GDC 61 with respect to appropriate containment, confinement, and filtering systems.

5.0 TECHNICAL ANALYSIS

The proposed changes would allow the containment equipment hatch to be open under administrative controls during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment, provided that it is capable of being closed. In addition the proposed changes would allow the emergency air lock to remain open during refueling activities under administrative controls similar to the personnel air lock. To support this plant scenario, elimination of automatic CPIS actuation during refueling, when the equipment hatch is open, ensures the availability of the containment purge system to maintain negative pressure in containment and to monitor any potential release via the purge exhaust system until the equipment hatch can be closed. Finally, to ensure the protection of control room by actuation of CRVIS, a new Surveillance Requirement is added to verify that the control room radiation monitor channels function within required time limits.

Allowing the equipment hatch to be open during CORE ALTERATIONS or movement of irradiated fuel raises the concern that radioactive materials could potentially be released through the open hatch and vented to the outside environment should accidents that involve fission product releases occur. Postulated accidents that could result in a release of radioactive material through the open hatch include a fuel handling accident that results in breaching of the fuel rod cladding, and a loss of residual heat removal (RHR) cooling event that leads to core boiling and uncovery. To provide the basis for justifying the proposed change, the concern with the potential radiological consequences of the two accidents that could result in a release of radioactive material through the open equipment hatch are discussed below.

Fuel Handling Accident (FHA)

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences are anticipated to result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel. Fuel handling accidents include dropping a single irradiated fuel assembly, or a handling tool or heavy object, onto other irradiated fuel assemblies.

The radiological consequences of a design basis fuel handling accident in containment have been previously evaluated assuming that the containment is open to the outside atmosphere. All airborne activity reaching the containment atmosphere is assumed to be exhausted to the environment within 2 hours of the accident.

Amendment No. 114 (Reference 8.1) approved leaving the containment air lock open during irradiated fuel movement and core alterations. In that application, AmerenUE recalculated the doses associated with a fuel handling accident. The analysis calculated the doses for the 0-2 hour period at the exclusion area boundary. The calculated doses were within the Standard Review Plan acceptance criteria of 6 REM to the whole body and 75 REM to the thyroid. As discussed in Amendment No. 114, the analysis assumes all radioactive material from the FHA is released to the environment within a two-hour period. This results in the same impact of potential dose consequences from a simultaneous release of gaseous effluents through an unisolated penetration flow path and the open personnel airlock doors (Reference 8.2). It also results in the same impact of potential dose consequences from a simultaneous release of gaseous effluents through an open containment equipment hatch. Therefore, allowing an open containment equipment hatch during core alterations or movement of irradiated fuel will not invalidate the conclusion that the potential dose consequences from a FHA will be well within the 10 CFR Part 100 limits.

In addition to the impact on offsite radiological consequences, the radiological consequences to the Control Room operator provided in support of Amendment No. 114 remain bounding. The radiological consequences to the Control Room operator continue to be bounded by the values reported in support of Amendment no.114. This conclusion is based on analyses that delays Control Room isolation to account for the new response times specified in the revised Technical Specification 3.3.7. The calculated offsite and control room operator doses are within the acceptance criteria of Standard Review Plan 15.7.4 (Reference 8.5) and General Design Criteria (GDC) 19.

On the basis of these evaluations, various revisions to Technical Specification Section 3.9.4, "Containment Penetrations" have been accepted by the NRC (References 8.1 and 8.2).

During refueling operations, the potential for containment pressurization as a result of a fuel handling accident is not likely. Therefore, the majority of the radioactive material releases from the accident would be held up inside containment with only a minimal amount of radioactive material released through the open equipment hatch. However, the combined dose consequences of this potential release with the releases through other unisolated penetration flow paths and the open personnel airlock doors, will be bounded by the current licensing basis fuel handling accident analysis. The current design basis fuel handling analysis does not credit the containment building barriers. It is assumed that all gap activity is released from the damaged rods and all the gaseous effluent escaping from the refueling pool is released directly to the environment within two hours through the open personnel airlock doors.

According to Section 15.7.4 of the Callaway FSAR (Reference 8.3), the resulting offsite dose consequences with both personnel air lock doors open were calculated to be 73.0 rem thyroid and 0.334 rem whole body at the exclusion area boundary. These results are well within the 10 CFR 100 limits. Since the total amount of radioactive material available for immediate releases into the water during a postulated fuel handling accident will be the same, the potential dose consequences from a simultaneous release of the gaseous effluents through the unisolated penetration flow paths, the open personnel airlock doors and the open equipment hatch will not be different from the previous analysis that assumes radioactivity to be released only through the open personnel airlock doors. Therefore, allowing the equipment hatch to be open during CORE ALTERATIONS or movement of irradiated fuel would not invalidate the conclusion that the potential dose consequences from a fuel handling accident will be well within the 10 CFR 100 guideline limits.

Loss of RHR Cooling

Release of radioactive materials are anticipated to be insignificant as a result of core boil-off due to a loss of RHR cooling, if the event does not continue for an extended period of time resulting in core uncovery and subsequent core damage. If core boil-off continues, the compartments in the vicinity of the core could be pressurized and thereby provide a driving force for the containment atmosphere to be released via the open hatch flow path to the outside atmosphere. However, the radiological consequences of this release of radioactive materials due to core boil-off, with no consideration for core uncovery and core damage, is expected to be significantly less than the radiological consequences arising from a postulated fuel handling accident because the total coolant activity (corresponding to a 1% fuel defect) is less than the total gap activities in the damaged rods at the earliest time fuel offloading may be commenced (100 hours after shutdown).

A review of calculations performed for the outage risk assessment revealed that the time to core boil would be greater than 5 hours should a loss of RHR cooling event occur at the beginning of fuel offloading, based on the normal water level maintained in the refueling pool (i.e., ≥ 23 ft above the top of the reactor vessel flange). Technical Specification 3.9.5 requires that corrective actions be taken immediately to restore the RHR cooling as soon as possible if RHR loop requirements are not met (by having one RHR loop operable and in operation). In addition, operators are required to close all containment penetrations providing direct access from the containment atmosphere to the outside environment within 4 hours. If an operator takes actions to restore the RHR cooling capability or uses an alternative method of core cooling within 5 hours time interval, the scenario involving core boiling and subsequent containment pressure pressurization would not be present. With all penetrations closed within the specified time period, the potential for the coolant to boil and subsequently release radioactive gas to the containment atmosphere, if the RHR cooling was not restored, would not be of concern.

Finally, CORE ALTERATIONS associated with activities other than those directly involving fuel movement have little likelihood of resulting in a fuel damage event involving an appreciable release of activity in containment. Additionally, a loss of RHR during these activities is considered unlikely given the short amount of time in this plant configuration and the consequences would be the same as described above.

Administrative Controls

NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 8.4), Section 11.3.6.5, provides the following guidance:

> ".... for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:
>
> - During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is

to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

The proposed changes do not affect the overall capability or availability for affected ventilation systems or radiation monitor detectors. The Control Room Emergency Ventilation System remains required to be OPERABLE by TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)" as well as the containment atmosphere radioactivity monitors (TS 3.3.6, "Containment Purge Isolation Instrumentation"). CPIS is changed from automatic actuation to manual actuation to provide assurance that any potential release is directed and monitored through the containment purge exhaust. CRVIS actuation remains automatic, however, in the event of a fuel handling accident in containment, the source of the signal is generated from the control room radiation monitor detectors rather than the containment purge system radiation monitor detectors. A new Surveillance Requirement is added to verify their ability to function. In the event that a fuel handling accident occurs in the fuel building, the fuel building ventilation radiation monitor detectors, GGRE0027 and GGRE0028, detect high radiation and initiate CRVIS. The affected containment penetrations that provide direct access to the outside atmosphere are the emergency airlock and the containment equipment hatch. The emergency air lock is maintained open under similar conditions as the personnel air lock. Existing TS requirements on other penetrations that provide direct access are not affected.

Containment ventilation at Callaway is accomplished via the Containment Purge and Exhaust System which includes the Containment Shutdown Purge System and Containment Minipurge System. These systems are not credited in any of the dose analyses, so there are no associated TS OPERABILITY requirements for these systems. The Containment Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating needed for prolonged containment access following a shutdown and during refueling. The system may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The Containment Minipurge System may be used during power operations to reduce the concentration of noble gases within the containment prior to and during personnel access or to equalize internal and external pressures. Both systems share purge supply and exhaust containment penetrations. Each penetration is equipped with two valves in parallel inside containment and two valves in parallel outside containment.

Once cold shutdown is achieved, only the Containment Shutdown Purge System is normally in operation. The system is manually initiated from the control room. The Containment Shutdown Purge System is designed to maintain the airborne radioactivity below the level required for personnel occupancy during refueling, and the Containment Minipurge System is designed to maintain airborne radioactivity below the required level for personnel occupancy during reactor power operation. The exhaust from these systems is ducted to the unit vent that is located at the top of the containment building. The HEPA filter elements and charcoal adsorber sections are tested periodically in accordance with Regulatory Guide 1.140. The handswitches for the fan units and the handswitches for the purge valves are located in the control room. Therefore, in the event of a fuel handling accident inside containment with the equipment hatch open, the containment purge can be easily controlled from the control room.

Exhaust from the containment is processed through the Containment Purge and Exhaust System charcoal adsorption train prior to discharge through the unit vent. The Containment Purge and Exhaust System is monitored for radioactivity, both upstream and downstream of the charcoal adsorber. The containment atmosphere radioactivity monitors (GT RE-31 and GT RE-32) continuously monitor the containment atmosphere for particulate, iodine, and gaseous radioactivity. The containment purge radiation monitors (GT RE-22 and GT RE-33) continuously monitor the containment purge exhaust duct during purge operations for particulate, iodine, and gaseous radioactivity. Normally these monitors would automatically isolate the Containment Purge and Exhaust System on high gaseous activity via the Engineered Safety Features Actuation System (ESFAS). However, the proposed revisions would eliminate this automatic function so that the purge system could remain in operation until the equipment hatch is closed. This would ensure negative pressure in containment and monitoring of a potential release. Although it is recognized that some exchange of air may occur at the open containment equipment hatch, the overall negative pressure in containment assures air flow through the containment purge exhaust system in compliance with GDC 64. In the event of a fuel handling accident inside containment, the control room alarm function of the required containment radiation monitors remains in service, and the radiation monitors provide indication of the magnitude of the release, thereby minimizing the potential for unmonitored release.

During CORE ALTERATIONS, Callaway FSAR Section 16.9.1.1 (Reference 8.5), requires that direct communications be maintained between the control room and personnel at the refueling station. Therefore, if a fuel handling accident were to occur inside containment, the control room would be immediately informed, and action would be promptly initiated in accordance with off-normal procedures to mitigate the consequences.

If open, the equipment hatch and the emergency air lock will be maintained in an isolable condition. The proposed revision to LCO 3.9.4 does not specifically include administrative control requirements for the emergency airlock and the containment equipment hatch. Unlike TS 3.6.3, Containment Isolation Valves, which include administrative controls in TS to ensure the status of multiple penetrations at higher MODES of operation, the administrative controls for the containment equipment hatch and the emergency airlock do not need to be specified in the TS. The TS and Bases provide the requirements for closure of the equipment hatch and the emergency air lock.

Administrative controls consist of written procedures and will be established prior to the implementation of the proposed changes. These procedural controls would require the following and would also be applicable to the emergency air lock:

1. Appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS.

2. Specified individuals are designated and readily available to close the equipment hatch following an evacuation that would occur in the event of a fuel handling accident.

3. Any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open equipment hatch can be quickly removed.

In addition new SR 3.9.4.2 ensures that the equipment necessary to close the hatch is at hand so that the hatch can be closed promptly in the event of a fuel handling accident inside containment. The equipment is dedicated for the purpose and the added surveillance precludes delays that would occur if the tools had to be rounded up. The same hardware, tools, equipment, and procedures are used to close the containment equipment hatch in all situations. As such there is no distinction between that which is required to close the hatch and that, which is required to close the hatch promptly. The added surveillance is sufficient for ensuring the necessary equipment is available and does not need to be duplicated as an administrative control.

When there is fuel in the reactor building and the containment equipment hatch is open, a designated individual will be present and available to direct closure of the containment equipment hatch. This is the same administrative control that is utilized to allow the personnel air lock to be open during CORE ALTERATIONS or movement of irradiated fuel assemblies (License Amendment No. 114). The same designated individual that is responsible for closing the air lock, following an evacuation that would occur in the event of a fuel handling accident, is also responsible for closing the equipment hatch. Direct and continuous communication with the control room is not necessary because the designated individual is readily available via other reliable communication systems. Training is provided to selected individuals responsible for various containment operations activities including personnel air lock, emergency air lock, and containment equipment hatch operation, as well as conditions that may require closure of these penetrations.

Given restated SR 3.9.4.2 to ensure dedicated tools and equipment and designated and trained individuals to close the equipment hatch, Callaway anticipates a typical closure time of less than one hour. This is based on past experience and discussions with containment coordinators. It is also typical of other licensee experiences such as Vogtle, the precedent plant. This time is well within the minimum time of 4 hours (TS 3.9.5, Required Action A.4) for the core to boil with loss of RHR cooling at the beginning of fuel offload.

Note, the emergency air lock closure capability is similar to the personnel air lock closure capability in that it is provided by the availability of at least one door and the same designated individual to close it. Under administrative controls, the containment equipment hatch would be isolated first followed by the personnel and emergency air locks.

The best estimate mission dose calculated for closing the containment equipment hatch following a postulated fuel handling accident is 24.6 rem thyroid. This calculation is based on an hour exposure time for personnel to install the equipment hatch.

Statements in this submittal concerning the backup diesel generator and the administrative controls for installing the missile barrier in the event of severe weather are contingency actions for abnormal events. These contingencies are addressed in plant procedures. The plant procedures are written with the intent that the equipment hatch be installed upon the arrival of threatening weather conditions that could generate missiles. Under severe weather conditions, the equipment hatch door is installed and the missile shield is positioned to provide adequate protection.

Callaway administrative controls are intended to be the same for the various containment openings. It is not necessary that the specific actions necessary for proper closure be contained in the TS Bases. Plant procedures address these items and changes to the procedures specifying the administrative controls fall within the 10 CFR 50.59 process. Making these administrative controls part of TS requirements or proposing a license condition would be inconsistent with what was approved in Amendment Nos. 115 and 93 for the Vogtle units and Amendment No. 114 for Callaway.

In conclusion, these administrative controls provide protection equivalent to that afforded by the administrative controls used to establish containment closure for a containment personnel air lock. An assessment of the radiological consequences, as described above for the proposed changes, concludes that site boundary doses remain well within the 10 CFR 100 limits and control room doses meet GDC 19 criteria without taking credit for closure of the equipment hatch. The administrative controls provide reasonable assurance that containment closure as a defense-in-depth measure can be reestablished quickly to limit releases much lower than assumed in the dose calculation.

Risk Significance

Based on the results of conservative dose calculations provided in this submittal, the risk to the health and safety of the public as a result of a fuel handling accident inside the containment with the equipment hatch open is minimal. Actual fuel handling accidents which have occurred in the past have resulted in minimal or no releases, which shows that the assumptions and methodology utilized in the radiological dose calculations are very conservative. Radioactive decay is a natural phenomenon. It has a reliability of 100 percent in reducing the radiological release from fuel bundles. In addition, the water level that covers the fuel bundles is another natural method that provides an adequate barrier to a significant radiological release. The requirement for at least 100 hours of decay prior to fuel movement is maintained in the Callaway FSAR Section 16.9.5 (Reference 8.6) and the requirement for water level is maintained in the TS. In addition, the requirements for isolable air locks, an isolable equipment hatch, isolable penetrations, and containment radiation monitors is maintained in the TS. The Containment Purge and Exhaust System will be available in accordance with the aforementioned NUMARC 93-01 guidelines to further reduce radiological release. Therefore, the risk to the health and safety of the public as a result of allowing the equipment hatch to be open during fuel movement is minimal.

6.0 REGULATORY ANALYSIS

The method of analysis used for evaluating the potential radiological consequences of the postulated fuel handling accident is in compliance with Regulatory Guide 1.25 and the guidance in NUREG-0800, Section 15.7.4 and NUREG/CR-5009. The analysis presented in Section 15.7.4 of the Callaway FSAR, demonstrating the adequacy of the system design features and plant procedures provided for the mitigation of the radiological consequences of postulated fuel handling accidents, assumes no credit is taken for iodine removal by the atmosphere filtration system filters. All radioactivity released to the containment is assumed to be released to the environment at ground level over a two hour period.

The technical analysis performed by AmerenUE demonstrates that the consequent doses at the exclusion area and low population zone boundaries are well within the limits of 10 CFR 100. Therefore, the proposed License amendment is in compliance with GDC 16, 56, 61, and 64 as well as Regulatory Guide 1.25, NUREG/CR-5009, and the criteria contained in NUREG-0800, Section 15.7.4.

In conclusion, based on the considerations discussed above, 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the Commission's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 NO SIGNIFICANT HAZARDS DETERMINATION

AmerenUE has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

The proposed changes will allow the equipment hatch to be open during CORE ALTERATIONS and movement of irradiated fuel assemblies inside containment. The status of the equipment hatch or the emergency air lock during refueling operations has no affect on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a fuel handling accident inside containment with an open equipment hatch are bounded by the current analysis described in the FSAR and the probability of an accident is not affected by the status of the equipment hatch, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analysis.

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

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

   Response: No

   The previously determined radiological dose consequences for a fuel handling accident inside containment with the air lock doors open remain bounding for the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR 100 and they meet the acceptance criteria of SRP section 15.7.4 and GDC 19.

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

Based on the above evaluations, AmerenUE concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

8.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. AmerenUE has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

As discussed above, the proposed changes do not involve a significant hazards consideration and the analysis demonstrates that the consequences from a fuel handling accident inside containment are well within the 10 CFR 100 limits. The implementation of administrative controls precludes a significant increase in occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

9.0 PRECEDENTS

There are precedents for allowing the equipment hatch to be open during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment. The Southern Nuclear Operating Company operating licenses for the Vogtle Generating Electric Plant Unit 1 and 2, have been amended to allow the equipment hatch to be open during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment. These amendments, Nos. 115 and 93, were issued on September 11, 2000.

10.0 REFERENCES

10.1 Letter dated July 15, 1996 from Kristine M. Thomas, NRC to Donald Schnell, Union Electric Company, "Callaway Plant - Amendment No. 114 to Facility Operating License No. NPF-30 (TAC No. M94456)."

10.2 Letter dated September 26, 2000 from Jack A. Donohew, NRC to Garry L. Randolph, Union Electric Company, "Callaway Plant - Amendment No. 138 to Facility Operating License No. NPF-30 (TAC No. MA9591)."

10.3 FSAR Section 15.7.4, "Fuel Handling Accidents."

10.4 NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," July 2000.

10.5 NUREG-0800, Standard Review Plan, Section 15.7.4, Rev. 1, July 1981.

10.6 FSAR Section 16.3, "Instrumentation."

10.7 FSAR Section 16.9, "Refueling Operations."

10.8 FSAR Section 16.11,"Offsite Dose Calculation Manual, Radioactive Effluent Controls."

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ATTACHMENT 3

MARKUP OF TECHNICAL SPECIFICATION PAGES

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------- NOTE------------- Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. ------------------------------------- One ~~or more Functions with one~~ or more manual channels ~~or automatic actuation trains~~ inoperable. ~~OR~~ ~~Both radiation monitoring channels inoperable.~~ ~~OR~~ ~~Required Action and associated Completion Time for Condition A not met.~~	C.1	Place and maintain containment purge supply and exhaust valves in closed position.	Immediately
	OR		
	C.2	Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust valves made inoperable by isolation instrumentation.	Immediately

Table 3.3.6-1 (page 1 of 1)
Containment Purge Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, (a), (b)	2	SR 3.3.6.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, ~~(a), (b)~~	2 trains	SR 3.3.6.2 SR 3.3.6.6	NA
3.	Containment Purge Exhaust Radiation - Gaseous	1, 2, 3, 4, ~~(a), (b)~~	2	SR 3.3.6.1 SR 3.3.6.3 SR 3.3.6.5	(c)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During CORE ALTERATIONS.
(b) During movement of irradiated fuel assemblies within containment.
(c) Set to ensure ODCM limits are not exceeded.

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days
SR 3.3.7.3	-----NOTE----- The continuity check may be excluded. ----- Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.7.4	-----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	18 months
SR 3.3.7.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.6	***-----NOTE-----*** ***Radiation monitor detectors are excluded from Response time testing.*** ***-----*** ***Verify Control Room Ventilation Isolation ESF RESPONSE TIMES are within limits.***	***18 months on a STAGGERED TEST BASIS***

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, 5, 6, and (a)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, 5, 6, and (a)	2 trains	SR 3.3.7.3	NA
		(c)	***2 trains***	***SR 3.3.7.6***	
3.	Control Room Radiation - Control Room Air Intakes	1, 2, 3, 4, 5, 6, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
		(c)	***2***	***SR 3.3.7.6***	
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			
5.	***Fuel Building Exhaust Radiation -Gaseous***	***Refer to LCO 3.3.8, "EES Actuation Instrumentation," for all initiation functions and requirements.***			

(a) During movement of irradiated fuel assemblies.

(b) Nominal Trip Setpoint concentration value ($\mu Ci/cm^3$) shall be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.

(c) During CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts, ***or if open, capable of being closed***;

b. One door in the emergency air lock ~~closed~~ and one door in the personnel air lock capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge Isolation valve.

------------------------------------ NOTE ------------------------------------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
~~SR 3.9.4.2~~	~~Verify each required containment purge isolation valve actuates to the isolation position on an actual or simulated actuation signal.~~	~~18 months~~
SR 3.9.4.2	***-----NOTE-----*** ***Only required for an open equipment hatch.*** ***----------*** ***Verify the capability to install the equipment hatch.***	***7 days***
SR 3.9.4.3	***Verify each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.***	***18 months***

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ATTACHMENT 4

RETYPED MARKUP OF TECHNICAL SPECIFICATION PAGES

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------- NOTE------------- Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. ------------------------------------- One or more manual channels inoperable.	C.1	Place and maintain containment purge supply and exhaust valves in closed position.	Immediately
	OR		
	C.2	Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust valves made inoperable by isolation instrumentation.	Immediately

Table 3.3.6-1 (page 1 of 1)
Containment Purge Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, (a), (b)	2	SR 3.3.6.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4,	2 trains	SR 3.3.6.2 SR 3.3.6.6	NA
3.	Containment Purge Exhaust Radiation - Gaseous	1, 2, 3, 4,	2	SR 3.3.6.1 SR 3.3.6.3 SR 3.3.6.5	(c)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During CORE ALTERATIONS.
(b) During movement of irradiated fuel assemblies within containment.
(c) Set to ensure ODCM limits are not exceeded.

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days
SR 3.3.7.3	---------NOTE--------- The continuity check may be excluded. ---------------------- Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.7.4	---------NOTE--------- Verification of setpoint is not required. ---------------------- Perform TADOT.	18 months
SR 3.3.7.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.6	---------**NOTE**--------- **Radiation monitor detectors are excluded from Response time testing.** ---------------------- **Verify Control Room Ventilation Isolation ESF RESPONSE TIMES are within limits.**	**18 months on a STAGGERED TEST BASIS**

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, 5, 6, and (a)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, 5, 6, and (a)	2 trains	SR 3.3.7.3	NA
		(c)	**2 trains**	**SR 3.3.7.6**	
3.	Control Room Radiation - Control Room Air Intakes	1, 2, 3, 4, 5, 6, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
		(c)	**2**	**SR 3.3.7.6**	
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			
5.	**Fuel Building Exhaust Radiation -Gaseous**	**Refer to LCO 3.3.8, "EES Actuation Instrumentation," for all initiation functions and requirements.**			

(a) During movement of irradiated fuel assemblies.

(b) Nominal Trip Setpoint concentration value ($\mu Ci/cm^3$) shall be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.

(c) During CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment.

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts**, or if open, capable of being closed**;

b. One door in the emergency air lock and one door in the personnel air lock capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge Isolation valve.

------------------------------------ NOTE ------------------------------------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	**-----NOTE-----** **Only required for an open equipment hatch.** **----------** **Verify the capability to install the equipment hatch.**	**7 days**
SR 3.9.4.3	**Verify each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.**	**18 months**

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ATTACHMENT 5

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

(for information only)

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge Isolation Instrumentation

BASES

BACKGROUND

Containment purge isolation instrumentation closes the containment isolation valves in the Mini-purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini-purge System may be in use during reactor operation and the Shutdown Purge System will be in use with the reactor shutdown.

Containment purge isolation initiates on an automatic or manual safety injection (SI) signal through the Containment Isolation - Phase A Function, or by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Two *gaseous* radiation monitoring channels are also provided as input to the containment purge isolation. The two channels measure gaseous radiation in a sample of the containment purge exhaust. Since the purge exhaust monitors constitute a sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from either of the two radiation monitoring channels initiates containment purge isolation, which closes both inner and outer containment isolation valves in the Mini-purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge isolation *gaseous* radiation ~~monitors~~ *channels* act as backup to the Phase A isolation signal to ensure closing of the purge supply and exhaust valves. ~~They are also the means for automatically isolating containment in the event of a fuel handling accident during shutdown; however, the dose calculations performed in support of Reference 5 do not assume automatic isolation (see also the Bases for LCO 3.9.4, "Containment Penetrations").~~ ***In the postulated fuel handling accident, the dose calculations performed in support of***

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(continued)

Reference 5 (open personnel airlock doors during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment) do not assume automatic containment purge isolation (see also the Bases for LCO 3.9.4, "Containment Penetrations").

Containment isolation ~~in turn~~ ensures

TSB CN 01-005

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment purge isolation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two push buttons in the control room.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet~~.~~ ***as well as the BOP ESFAS output actuation relays needed to effect a manual containment purge isolation.*** TSB CN 01-005

2. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of containment purge isolation.

Automatic Actuation Logic and Actuation Relays (BOP ESFAS) consist of the same features and operate in the same manner as described for ESFAS Function 6.c, Auxiliary Feedwater.

3. Containment Purge Exhaust Radiation - *Gaseous*

The LCO specifies two required *Containment Purge Exhaust Radiation – Gaseous* channels ~~of radiation monitors~~ (GTRE0022 and GTRE0033) to ensure that the radiation monitoring

(continued)

BASES

LCO

3. Containment Purge Exhaust Radiation (continued)

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY also requires correct valve lineups and sample pump operation, as well as detector OPERABILITY, since these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

4. Containment Isolation - Phase A

Containment Purge Isolation is also initiated by all Table 3.3.2-1 Functions that initiate Containment Isolation - Phase A. Therefore, the requirements are not repeated in Table 3.3.6-1. Instead, refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), Containment Isolation - Phase A, and Containment Purge Exhaust Radiation ~~- Gaseous Functions are~~ required OPERABLE in MODES 1, 2, 3, and 4. The Containment Purge ~~Exhaust Radiation - Gaseous,~~ Manual Initiation ***Function***~~, and BOP ESFAS Logic Functions are~~ ***is*** also required OPERABLE during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. ***During CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment, automatic actuation functions of the containment purge isolation gaseous radiation channels are not required to be OPERABLE.***

TSB CN 01-005

The automatic actuation logic and actuation relays for the Containment Purge Exhaust Radiation - Gaseous channels (GTRE0022 and GTRE0033) are not required to be OPERABLE during CORE ALTERATIONS or during the movement of irradiated fuel assemblies within containment, except for those BOP ESFAS output actuation relays needed to effect a manual containment purge isolation. If required, containment purge isolation can be initiated manually from the control room.

Under these conditions ***and MODES 1,2,3,4, and other conditions discussed above***, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge isolation instrumentation must be OPERABLE in these MODES.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

TSB CN 01-005

Condition C applies to the Manual Initiation ***Function.*** ~~, Automatic BOP ESFAS Actuation Logic and Actuation Relays, and Containment Purge Exhaust Radiation - *Gaseous* Functions and addresses the train orientation of the BOP ESFAS. It also addresses the failure of both *gaseous* radiation monitoring channels, or the inability to restore a single failed *gaseous* radiation monitoring channel to OPERABLE status in the time allowed for Required Action A.1.~~ If one or more ~~BOP ESFAS logic trains or~~ manual initiation channels are inoperable~~, both *gaseous* radiation monitoring channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met,~~ ***then*** operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are ~~met for each valve made~~ inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Purge Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

means to ensure control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident. The probability of a waste gas decay tank rupture accident occurring during the period of time outside the Applicability (i.e., not in MODES 1-6 and with no movement of irradiated fuel assemblies occurring) is insignificant. There are no safety analyses that take credit for CREVS actuation upon ~~an FBVIS or~~a high containment purge exhaust radiation. **A FBVIS is credited to protect the control room in the event of a design basis fuel handling accident inside the fuel building.**

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Sources of CRVIS initiation which are remote from the Control Room intake louvers are not response time tested. For example, GGRE0027 and GGRE0028, which monitor Fuel Building exhaust are not response time tested. The analysis does credit a Fuel Building Isolation Signal (FBVIS) for actuating a CRVIS following a Fuel Handling Accident in the Fuel Building. Due to the remote location of the Fuel Building radiation monitors relative to the Control Room intake louvers, the FBVIS will isolate the Control Room prior to the post-accident radioactive plume reaching the Control Room intake louvers.

Similarly, for a LOCA, the analysis credits a time zero Control Room isolation. A Safety Injection signal initiates a Containment Isolation Phase A, which initiates a CRVIS. This function is also credited for isolating the Control Room prior to the post-accident radioactive plume reaching the Control Room intake louvers.

For a Fuel Handling Accident within Containment, GKRE0004 and GKRE0005 are credited for initiating a CRVIS. These monitors are not remote from the Control Room intake louvers. They are downstream of the Control Room intake. Therefore, a specific response time is modeled, and a response time Surveillance Requirement is imposed for this CRVIS function.

The CREVS actuation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREVS is OPERABLE.

1. Manual Initiation

(continued)

BASES

LCO

3. Control Room Radiation – Control Room Air Intake (continued)

instrumentation necessary to initiate the CREVS remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY also requires correct valve lineups and sample pump operation, as well as detector OPERABILITY, since these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses. The required radiation monitor's' OPERABILITY is not dependent on forced flow in the control room supply duct. GKRE0004 and GKRE0005 OPERABILITY is not dependent on the status of GKHZ0013D/0057A/0150/0151, SGK02, or CGK01A and B. GKRE0004 and GKRE0005 may be considered OPERABLE with CREVS in the CRVIS mode of operation.

4. Containment Isolation - Phase A

Control Room Ventilation Isolation is also initiated by all Table 3.3.2-1 Functions that initiate Containment Isolation - Phase A. Therefore, the requirements are not repeated in Table 3.3.7-1. Instead, refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

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5. Fuel Building Exhaust Radiation-Gaseous

During movement of irradiated fuel assemblies in the fuel building, Control Room Ventilation Isolation is initiated by high radiation in the fuel building detected by fuel building radiation monitor detectors GGRE0027 and GGRE0028. The requirements are not repeated in Table 3.3.7-1. Instead, refer to LCO 3.3.8 for all initiating Functions and requirements.

APPLICABILITY

All CREVS Functions, including actuation on the Containment Isolation - Phase A Function, must be OPERABLE in MODES 1, 2, 3, and 4. The Manual Initiation, Automatic Actuation Logic and Actuation Relays (BOP ESFAS), and Control Room Radiation – Control Room Air Intake Functions are also required OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies. These Functions must be OPERABLE in MODES 5 and 6 for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators. **During CORE ALTERATIONS or during movement of irradiated fuel assemblies in containment, the Control Room**

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(continued)

BASES

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Radiation monitors (GKRE004 and GKRE005) also assure the generation of a control room ventilation isolation signal (CRVIS) on detection of high gaseous activity in the event of a fuel handling accident within containment.

During movement of irradiated fuel assemblies in the fuel building, the fuel building radiation monitors (GGRE0027 and GGRE0028) assure the generation of a control room ventilation isolation signal (CRVIS) on detection of high gaseous activity in the event of a fuel handling accident in the fuel building.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. It the measured Trip Setpoint is less conservative than the tolerance specified by the calibration

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.7.4 (continued)

operating experience. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The channels tested have no setpoints associated with them.

SR 3.3.7.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.7.6

SR 3.3.7.6 is the performance of the required response time verification every 18 months on a STAGGERED TEST BASIS on those functions with time limits provided in Reference 2. Each verification shall include at least one train such that both trains are verified at least once per 36 months.

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SR 3.3.7.6 is modified by a Note stating that the radiation monitor detectors are excluded from ESF RESPONSE TIME testing. The Note is necessary because of the difficulty associated with generating an appropriate radiation monitor detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response. Response time of the channel shall be verified from the detector output or input to the first electronic component in the channel.

REFERENCES

1. FSAR Section 7.3.4 and Table 7.3-8.
2. **FSAR Table 16.3-2.**

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50 Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment and ***if closed, the containment*** equipment hatch must be held in place by at least four bolts. ***Alternatively, the equipment hatch can be open provided it can be installed with a minimum of four bolts holding it in place.*** Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

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The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." ***The personnel air lock is nominally a right circular cylinder, approximately 10 ft in diameter with a door at each end. The emergency air lock is approximately 5 ft 9 in inside diameter with a 2 ft 6 in door at each end.*** ~~Each air lock has a door at both ends.~~ The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door

(continued)

interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within

(continued)

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BASES

BACKGROUND (continued)

containment, containment closure is required ***under administrative controls***. The door interlock mechanism may remain disabled; however, one personnel air lock door ~~must be capable of being closed~~ and one emergency air lock door must be ~~closed~~***capable of being closed***.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge System includes two subsystems. The Shutdown Purge subsystem includes a 36 inch supply penetration and a 36 inch exhaust penetration. The second subsystem, a minipurge system, includes an 18 inch supply penetration and an 18 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the Shutdown Purge supply and exhaust penetrations are secured in the closed position or blind flanged. The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5 or MODE 6 excluding CORE ALTERATIONS or movement of irradiated fuel in containment.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The Shutdown purge system is used for this purpose, and all four valves are capable of being closed by the ESFAS in accordance with LCO 3.3.6, "Containment Purge Isolation Instrumentation," during CORE ALTERATIONS or movement of irradiated fuel in containment.

Typically the minipurge system is not used in MODE 6.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier (such as a valve, flange, or penetration sealing mechanism) for the other containment penetrations during fuel movements.

"Direct access from the containment atmosphere" is defined as: The

(continued)

BASES

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). The fuel handling accident (in containment) analyzed in Reference 2 consists of dropping a single irradiated fuel assembly onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Pool Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values.

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits ~~the consequences of a fuel handling accident in containment~~ by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge penetrations ~~and the personnel air lock.~~ ***and the personnel air lock, the emergency air lock, and the equipment hatch, which must be capable of being closed.*** For the OPERABLE containment purge penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge Isolation System to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit. ***During CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment, Containment Purge Isolation valves are OPERABLE if they are capable of being closed by manual actuation.*** For the containment personnel air lock ***and the emergency air lock,*** one air lock door must be capable of being closed. ***Thus b***~~B~~oth containment personnel air lock ***and emergency air lock*** doors may be open during movement of irradiated fuel or CORE ALTERATIONS, provided an air lock door ***for each air lock*** is capable of being closed. Administrative controls ensure that 1) appropriate personnel are aware that both personnel air lock ***and emergency air lock*** doors are open, 2) a specified individual(s) is designated and available to close the air lock~~*(s)*~~ following a required evacuation of containment, and 3) any obstruction(s) (e.g. cables and hoses) that could prevent closure of an open air lock can be quickly removed (Ref. 1).

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(continued)

BASES

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The equipment hatch may be open during movement of irradiated fuel or CORE ALTERATIONS provided the hatch is capable of being closed. Administrative controls include 1) appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS, 2) specified individuals are designated and readily available to close the equipment hatch following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of the equipment hatch can be quickly removed. Administrative controls also ensure that during CORE ALTERATIONS and/or during the movement of irradiated fuel assemblies within containment and when the containment equipment hatch is open, either the minipurge or the shutdown purge exhaust system is in service; the trip setpoint function for the purge radiation monitor detectors GTRE0022 and GTRE0033 is bypassed; and the requirements of TS LCO 3.3.7, CREVS Actuation Instrumentation, are met.

The LCO is modified by a NOTE allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and

LCO (continued)

2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident (Ref. 4).

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. Proper installation and removal of the upper internals with irradiated fuel in the reactor vessel does not constitute a CORE ALTERATION or a movement of irradiated fuel. Therefore, this LCO is not applicable during installation and removal of the reactor vessel upper internals.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the outside atmosphere.

SR 3.9.4.2

This Surveillance demonstrates that the necessary hardware, tools, and equipment are available to install the equipment hatch. The equipment hatch is provided with a set of hardware, tools, and equipment for moving the hatch from its storage location and installing it in the opening. The required set of hardware, tools, and equipment shall be inspected to ensure that they can perform the required functions.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of the time to complete the fuel handling operations. The Surveillance is modified by a Note that only requires that the Surveillance be met for an open equipment hatch. If the equipment hatch is installed in its opening, the availability of the means to install the hatch is not required. The 7 day Frequency is adequate considering that the hardware, tools, and equipment are dedicated to the equipment hatch and not used for any other function.

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SR 3.9.4.~~2~~***3***

This Surveillance demonstrates that each containment purge isolation valve actuates to its isolation position on manual initiation.~~or on an actual or simulated high radiation signal.~~ The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge Isolation instrumentation requires a CHANNEL CHECK every 12 hours, an ACTUATION LOGIC TEST every 31 days on a STAGGERED TEST~~S~~ BASIS, and a COT every 92 days to ensure the channel OPERABILITY during~~refueling operations~~ ***MODES 1,2,3,and 4.*** Every 18 months a TADOT and a CHANNEL CALIBRATION are performed. The system actuation response time is demonstrated every 18 months on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances will ensure that the valves are capable of ~~closing~~ ***being manually closed*** after a postulated fuel

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ATTACHMENT 6

PROPOSED FSAR CHANGES

(for information only)

TABLE 11.5-3

AIRBORNE PROCESS RADIOACTIVITY MONITORS,

Monitor	Type (continuous)	Range (μCi/cc)	MDC (1) (μCi/cc)	Controlling Isotope	Alert (16) Alarm (μCi/cc)	Hi (16) Alarm (μCi/cc)	Total Ventilation Flow (cfm)	Minimum Required Sensitivity (μCi/cc)	Monitor Control Function
0-GT-RE-31 0-GT-RE-32 Containment atmosphere monitors	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1×10^{-11} 1×10^{-10} 2×10^{-7}	Cs-137 I-131 Xe-133	1.0×10^{-8} 1.0×10^{-8} 3.0×10^{-4}	1.0×10^{-7} 9.0×10^{-7} 6.0×10^{-4}	420,000 420,000 420,000	1×10^{-7} (7) 9×10^{-8} (7) 1×10^{-4} (7)	NA
0-GT-RE-22 0-GT-RE-33 Containment purge system monitors	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1×10^{-11} 1×10^{-10} 2×10^{-7}	Cs-137 I-131 Xe-133	5.0×10^{-8} 5.0×10^{-8} (12)	1.0×10^{-7} 9.0×10^{-8} (11) (15)	20,000/4000 20,000/4000 20,000/4000	1×10^{-7} (7) 9×10^{-8} (7) 1×10^{-4} (7)	Isolates containment purge, deenergizes purge fans on high gaseous activity via the ESFAS (see Section 7.3)
O-GT-RE-59 O-GT-RE-60 Containment high activity monitors	Gamma (5)	1 to 10^{8} $\frac{\text{rads}}{\text{hr}}$	1 $\frac{\text{rad}}{\text{hr}}$	NA	1.6×10^{3} R/hr	1.6×10^{4} R/hr	NA	NA	NA
O-GE-RE-92 Condenser air discharge monitor	Gaseous (continuous) (3), (6)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	2×10^{-6} (9)	2×10^{-5} (10)	25	NA	Closes blowdown isolation valve on Hi alarms

See CN #80-065

TABLE 11.5-3 (Sheet 2)

AIRBORNE PROCESS RADIOACTIVITY MONITORS

Monitor	Type (continuous)	Range (µCi/cc)	MDC (1) (µCi/cc)	Controlling Isotope	Alert (16) Alarm (µCi/cc)	Hi (16) Alarm (µCi/cc)	Total Ventilation Flow (cfm)	Minimum Required Sensitivity (µCi/cc)	Monitor Control Function
O-GG-RE-27 O-GG-RE-28 Fuel building exhaust monitors (2)	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1×10^{-11} 1×10^{-10} 2×10^{-7}	Cs-137 I-131 Xe-133	1×10^{-8} (8) 9×10^{-9} (8) 1.6×10^{-3}	1×10^{-7} (7) 9×10^{-8} (7) 3.2×10^{-3} (14)	20,000 20,000 20,000	1×10^{-7} (7) 9×10^{-8} (7) 1×10^{-4} (7)	Initiates switch to fuel building emergency ventilation on high gaseous activity via the ESFAS (see Section 7.3)
O-GK-RE-04 O-GK-RE-05 Control room air supply monitors	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1×10^{-11} 1×10^{-10} 2×10^{-7}	Cs-137 I-131 Xe-133	1×10^{-8} (8) 9×10^{-9} (8) 1.1×10^{-3}	1×10^{-7} (7) 9×10^{-8} (7) 2.2×10^{-3} (13)	2000 2000 2000	1×10^{-7} (7) 9×10^{-8} (7) 1×10^{-4} (7)	Initiates switch to control room emergency ventilation on high gaseous activity via the ESFAS (see Section 7.3)

Sample flow for each channel is 3 cfm

(1) MDC = minimum detectable concentration.
(2) When fuel is in the building.
(3) Beta scintillation detector.
(4) Gamma scintillation detector.
(5) Gamma sensitive ion chamber.
(6) When in operation.
(7) 10 MPC.
(8) MPC
(9) One order of magnitude above MDC to avoid spurious alarms and to indicate primary to secondary leakage.
(10) Two orders of magnitude above MDC to indicate significant inleakage of radioactivity.
(11) High alarm is set to ensure that Offsite Dose Calculation Manual limits are not exceeded.
(12) Alert alarm is administratively established at a point sufficiently below the High alarm so as to provide additional assurance that Offsite Dose Calculation Manual limits are not exceeded.
(13) Submersion dose rate does not exceed 2 mr/hr in the control room.
(14) Submersion dose rate does not exceed 4 mr/hr in the fuel building.
(15) High alarm setpoint is established to ensure that Offsite Dose Calculation Manual limits are not exceeded ~~and is limited to 5 x 10^-3 µci/cc during core alterations or movement of irradiated fuel within the containment.~~
(16) Alert and High alarm values do not include instrument loop uncertainty estimates.

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See CN#96-04

TABLE 12.3-3 Table 12.3-3

INPLANT AIRBORNE RADIOACTIVITY MONITORS

Monitor	Type (continuous)	Range µCi/cc	MDC(1) µCi/cc	Controlling Isotope	Alert (15) Alarm µCi/cc	
OGTRE31	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
OGTRE32	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	1.0×10^{-8}	
Containment atmosphere monitors	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	3.0×10^{-4}	
OGTRE22	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	5×10^{-8}	
OGTRE33	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	5×10^{-8}	
Containment purge system monitors	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	(10)	
OGGRE27	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
OGGRE28	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	9×10^{-9}	
Fuel building exhaust monitors (2)	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	1.6×10^{-3}	
OGKRE04	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
OGKRE05	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	9×10^{-9}	
Control room air supply monitors	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	1.1×10^{-3}	
OGLRE60 Auxiliary Building ventilation exhaust monitor	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
OGKRE41 Access control area ventilation exhaust monitor	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-9}	(8)
OGHRE23 Waste gas decay tank area ventilation exhaust monitor	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Kr85 Xe-133	1×10^{-5}	(8)
Portable monitor	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	NA	
	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	NA	
	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Kr85	NA	

TABLE 12.3-3 (Sheet 2)

INPLANT AIRBORNE RADIOACTIVITY MONITORS

High (15) Alarm µCi/cc	Flow Ventilation Flow (cfm)	Subcompartment Flow Rate (cfm)	Dilution Factor	Minimum Required Sensitivity (µCi/cc)	Monitor Control Function
1×10^{-7}	420,000	NA	NA	1×10^{-7} (6)	NA
9×10^{-7}	420,000	NA	NA	9×10^{-8} (6)	
6.0×10^{-4}	420,000	NA	NA	1×10^{-4} (6)	
1×10^{-7}	20,000/4,000	NA	NA	1×10^{-7} (6)	See Table 11.5-3 for process and control functions.
9×10^{-8}	20,000/4,000	NA	NA	9×10^{-8} (6)	
(11)	20,000/4,000	NA	NA	1×10^{-4} (6)	
(14)					
1×10^{-7}	20,000	NA	NA	1×10^{-7} (6)	See Table 11.5-3 for process control functions.
9×10^{-8}	20,000	NA	NA	9×10^{-8} (6)	
3.2×10^{-3} (13)	20,000	NA	NA	1×10^{-4} (6)	
1×10^{-7}	2000	NA	NA	1×10^{-7} (6)	See Table 11.5-3 for process control functions.
9×10^{-8}	2000	NA	NA	9×10^{-8} (6)	
2.2×10^{-3} (12)	2000	NA	NA	1×10^{-4} (6)	
1×10^{-7}	12,000	100	8×10^{-3} (5)	8×10^{-10} (6),(9)	Alarms
1×10^{-8} (7)	6,000	100	1.67×10^{-2} (5)	1.67×10^{-9} (6),(9)	Alarms
1×10^{-4} (7)	500	250	0.5 (5)	5×10^{-5} (6),(9)	Alarms
NA	NA	NA	NA		Alarms
NA	NA	NA	NA		
NA	NA	NA	NA		

TABLE 12.3-3 (Sheet 3)

INPLANT AIRBORNE RADIOACTIVITY MONITORS (CALLAWAY)

Sample Flow for each channel is 3 cfm.

(1) MDC = minimum detectable concentration.

(2) When fuel is in the building.

(3) Beta scintillation detector.

(4) Gamma scintillation detector.

(5) Dilution factor = $\frac{\text{Subcompartmental flow in cfm}}{\text{Total flow in cfm}}$

(6) Minimum required sensitivity of monitor in µCi/cc at 10 MPChrs for the controlling isotope = dilution factor x 10 MPC.

(7) 10 MPC x dilution factor.

(8) MPC x dilution factor.

(9) Grab samples will be analyzed in the laboratory, and iodine concentrations will be calculated, using previously established ratios.

(10) Alert alarm is administratively established at a point sufficiently below the High alarm so as to provide additional assurance that Offsite Dose Calculation Manual (ODCM) limits are not exceeded.

(11) High alarm is set to ensure that ODCM limits are not exceeded.

(12) Submersion dose rate does not exceed 2 mr/hr in the control room.

(13) Submersion dose rate does not exceed 4 mr/hr in the fuel building.

(14) High alarm setpoint is established to ensure that ODCM limits are not exceeded ~~and is limited to 5 x 10[illegible] µCi/cc during core alterations or movement of irradiated fuel within the containment.~~

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(15) Alert and High alarm setpoint values do not include instrument loop uncertainty estimates.

TABLE 16.3-2 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES(11)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
9. Steam Generator Water Level-Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60(8)
b. Start Turbine-Driven Auxiliary Feedwater Pump	≤ 60(8)
c. Feedwater Isolation	≤ 2(5),(8)
10. Loss-of-Offsite Power	
Start Turbine-Driven Auxiliary Feedwater Pump	≤ 60(9)
11. Trip of All Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	N.A.
12. Auxiliary Feedwater Pump Suction Pressure-Low	
Transfer to Essential Service Water	≤ 60(1)
13. RWST Level-Low-Low Coincident with Safety Injection	
Automatic Switchover to Containment Sump	≤ 40(10)
14. Loss of Power	
a. 4 kV Bus Undervoltage-Loss of Voltage	≤ 14(6)
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	≤ 144(13)
15. Phase "A" Isolation	
a. Control Room Isolation	N.A.
b. Containment Purge Isolation	≤ 2(5)
16. Control Room High Gaseous Activity	
Control Room Isolation	≤ 60(14)

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TABLE 16.3-2 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

TABLE NOTATIONS

(11) NRC approved the use of allocated response times for some components in their letter from Jack Donohew to Garry L. Randolph, "Application of WCAP-14036-P-A for Response Time Testing Elimination at Callaway Plant, Unit 1 (TAC NO. MA7283)," dated March 3, 2000.

(12) Signal actuation and valve stroke time delays included.

(13) Signal actuation, diesel generator starting, loss of voltage time delay relay, LSELS degraded voltage bistable delay timers and degraded voltage feeder breaker time delay relays included.

Insert 4a

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INSERT 4a

(14) The radiation monitor detector is excluded from response time testing. The stated response time accounts for the elapsed time between introduction of a count rate from the detector corresponding to the actuation setpoint and repositioning of the components necessary to achieve Control Room isolation.

16.11.2.4 RADIOACTIVE GASEOUS EFFLUENT MONITORING
(3/4.3.3.10) INSTRUMENTATION LIMITING CONDITION FOR OPERATION

(ODCM 9.2.1)

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 16.11-5 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Section 16.11.2.1 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Section 16.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 16.11-5.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 16.11-5. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Radioactive Effluent Release Report, pursuant to Technical Specification 5.6.3, why this inoperability was not corrected within the time specified.

c. The provisions of Sections 16.0.1.3 and 16.0.1.4 are not applicable.

16.11.2.4.1 SURVEILLANCE REQUIREMENTS

(ODCM 9.2.2)

a. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL OPERATIONAL TEST at the frequencies shown in Table 16.11-6.

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~~b. Verify the trip setpoint/concentration value for Containment Purge Monitors (GT-RE-22 and GT-RE-33) is set at less than or equal to 5E-3μCi/cc during CORE ALTERATIONS or movement of irradiated fuel within the containment.~~

16.11.2.4.2 BASES

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

Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Section 16.11.2.1 shall be such that concentrations as low as 1 x 10^{-6} µCi/cc are measurable.

~~The restriction on the setpoint for GT-RE-22 and GT-RE-33 is based on a fuel handling accident inside the Containment Building with resulting damage to one fuel rod and subsequent release of 0.1% of the noble gas rod activity, except for 0.3% of the Kr-85 activity. The setpoint concentration of 5E-3µCi/cc is is equivalent to approximately 150 mR/hr submersion dose rate.~~

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TABLE 16.11-5

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Unit Vent System			
a. Noble Gas Activity Monitor - Providing Alarm (GT-RE-21)	1	At all times	40,46
b. Iodine Sampler	1	At all times	43
c. Particulate Sampler	1	At all times	43
d. Unit Vent Flow Rate	1	At all times	45
e. Particulate and Radioiodine Sampler Flow Rate Monitor	1	At all times	~~39~~ 43 BPH 05/01/00
2. Containment Purge System		alarm functions only	FSAR CN 01-030
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (GT-RE-22, GT-RE-33)	2	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	41
b. Iodine Sampler	1	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	43
c. Particulate Sampler	1	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	43
d. Containment Purge Ventilation Flow Rate	N/A	N/A	N/A
e. Particulate and Radioiodine Sampler Flow Rate Monitor	1	MODES 1,2,3,4 and during CORE ALTERATIONS or movement of irradiated fuel within the containment	~~39~~ 43 BPH 05/01/00
3. Radwaste Building Vent System			
a. Noble Gas Activity Monitor-Providing Alarm and Automatic Termination of Release (GH-RE-10)	1	At all times	38,40

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ATTACHMENT 7

SUMMARY OF REGULATORY COMMITMENTS

SUMMARY OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by AmerenUE, Callaway Plant in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Dave E. Shafer, Superintendent, Licensing at AmerenUE, Callaway Plant, (314) 554-3104.

COMMITMENT	Due Date/Event
The amendment for allowing the containment equipment hatch and the emergency airlock to be open during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies will be implemented prior to Refueling Outage 12.	Prior to MODE 6 of Refueling Outage 12
Administrative controls consisting of written procedures will be established prior to the implementation of the proposed change. These procedural controls would require: 1) appropriate personnel are aware of the open status of the containment during movement of irradiated fuel or CORE ALTERATIONS, 2) specified individuals are designated and readily available to close the containment equipment hatch and emergency airlock following an evacuation that would occur in the event of a fuel handling accident, and 3) any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open equipment hatch or emergency airlock can be quickly removed.	Prior to MODE 6 of Refueling Outage 12
Administrative controls consisting of written procedures will be established to ensure that during CORE ALTERATIONS and/or during movement of irradiated fuel assemblies within containment and when the containment equipment hatch is open, either the minipurge or the shutdown purge exhaust system is in service; the trip setpoint function for the purge radiation monitor detectors GTRE0022 and GTRE0033 is bypassed; and the requirements of TS LCO 3.3.7, CREVS Actuation Instrumentation, are met.	Prior to MODE 6 of Refueling Outage 12