June 16, 2011

ULNRC-05793

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

10 CFR 50.90

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC CO.
APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE NPF-30 (LDCN 10-0036)
REVISION OF TECHNICAL SPECIFICATION 3.3.8, TAC NO. ME5173

Reference: Ameren Missouri letter ULNRC-05744 dated December 10, 2010

In the referenced document above, Ameren Missouri submitted an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant. That amendment application proposed changes to Technical Specification (TS) 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation," that would add new Surveillance Requirement (SR) 3.3.8.6. The new SR would require the performance of response time testing on the portion of the EES required to isolate the normal fuel building ventilation exhaust flow path and initiate the fuel building ventilation isolation signal (FBVIS) mode of operation.

During the NRC staff's review a request for additional information (RAI) was identified and issued electronically on May 19, 2011 with a response requested by June 17, 2011. Attachment 1 provides the requested information. The information provided in Attachment 1 does not affect the licensing evaluations submitted in the referenced amendment application nor does Attachment 1 alter the conclusions of those licensing evaluations. Attachment 2 provides pages from Callaway's Final Safety Analysis Report (FSAR) (some with conforming markups) that discuss fuel handling accident (FHA) analysis assumptions. Attachment 3 provides a TS 3.3.7 Bases markup for clarification of the information-only Bases markups included in the referenced amendment application.

Ameren Missouri continues to request approval of this license amendment request prior to October 1, 2011 so that it can be implemented before the next refueling outage (Refuel 18, October 2011). Ameren Missouri further requests that the license amendment be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance with the following exception:

Since SR 3.3.8.6 is a new Surveillance Requirement, the first required performance will come due by the end of the first surveillance interval that begins or is in effect on the date of implementation of this amendment. (This is similar to the License Condition applied to new Surveillance Requirements added by License Amendment 133 for the ITS Conversion.) As such, if the license amendment is issued prior to October 1, 2011, SR 3.3.8.6 will first be met during Refuel 18. If the license amendment is issued after October 1, 2011, SR 3.3.8.6 will first be met during Refuel 19 (spring 2013).

As was the case with the referenced amendment application, no commitments are contained in this correspondence. If you have any questions on this amendment application or the attached information, please contact me at (573) 676-8719 or Mr. Thomas Elwood at (314) 225-1905.

I declare under penalty of perjury that the foregoing and attached is true and correct.

Very truly yours,

Executed on: 6/16/2011

Scott A. M

Scott A. Maglio
Regulatory Affairs Manager

GGY/nls

Attachments:

1 – Response to NRC Request for Additional Information Regarding LDCN 10-0036
2 – FHA Analysis Assumptions and Associated FSAR Markups
3 – TS 3.3.7 Applicable Safety Analyses Bases Clarification

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ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
REGARDING LDCN 10-0036

REQUEST FOR ADDITIONAL INFORMATION
CALLAWAY PLANT, UNIT 1
UNION ELECTRIC COMPANY,
LICENSE AMENDMENT APPLICATION (LCDN 10-0036) TO ADD NEW SURVEILLANCE REQUIREMENT 3.3.8.6 TO TECHNICAL SPECIFICATION 3.3.8, "EMERGENCY EXHAUST SYSTEM ACTUATION INSTRUMENTATION"
TAC NUMBER ME5173

The NRC staff requests additional information to complete its review of the license amendment request (LAR) to add new Surveillance Requirement (SR) 3.3.8.6 to Technical Specification (TS) 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation."

By letter dated December 10, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML103470204), Union Electric Company (the licensee) proposed this new SR that would require the performance of response time testing on the portion of the EES required to isolate the normal fuel building ventilation exhaust flow path and initiate the fuel building ventilation isolation signal (FBVIS) mode of operation. By email dated January 14, 2011 (ADAMS Accession No. ML110140849), the NRC provided its acceptance of this amendment request which will enable the NRC to complete its detailed technical review.

From NRC staff review of the provided application, the NRC staff has the following questions listed below.

Question 1

In Attachment 1, page 4, the LAR proposed to change a parameter (assumption) used in evaluating the radiological consequences of a design basis Fuel Handling Accident (FHA) at Callaway Plant (Callaway). Specifically, a 90-second response time is proposed as an appropriate limit for Fuel Building Ventilation Exhaust engineered safety feature (ESF) response time. The LAR indicated that because the fuel building FHA analysis had not previously been performed with an assumed Fuel Building Ventilation Exhaust ESF response time of 90 seconds, reanalysis of this event resulted in small increases in the calculated dose consequences. The new/recalculated dose values for the exclusion area boundary (EAB) and low population zone (LPZ) are reflected on the mark-up for Table 15.7-8, which was provided in Attachment 5 of the LAR. However, no new/recalculated control room (CR) dose values for the proposed change are reported.

(a) Pursuant with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 19, "Control Room," please provide the new/recalculated dose to verify that the proposed change would not result in CR dose consequences that exceed the 5 rem regulatory limit.

(b) Please also verify that no other parameters, assumptions, and/or methodologies have been changed as it relates to the current radiological consequence analyses of record (AOR) for Callaway.

(c) In addition, please provide an evaluation of the impact of the proposed change on all applicable accidents in the design bases or include a justification supporting why an evaluation of the impact is not needed. If an evaluation of other design bases accidents is provided, please provide the regulatory bases for the acceptance criteria (e.g. 10 CFR, Part 50, Appendix A, GDC 19, 10 CFR Part 100, etc.) and any regulatory guidance used to make this determination.

(d) Additionally, please provide confirmation that the new/recalculated dose values will become the updated licensing basis values for the applicable accidents evaluated and will be included in the next revision to the Updated Final Safety Analysis Report (UFSAR) for Callaway.

RAI Responses

1.(a) Control room doses for a fuel handling accident in the fuel building are as follows:

- 0.753 rem thyroid
- 0.009 rem whole body
- 0.796 rem beta skin.

The calculation for the above values assumes that control room isolation occurs prior to radioactivity entering the normal control room ventilation air intake such that the control room operators do not experience unfiltered ventilation flow other than in-leakage as depicted in FSAR Figure 15A-2. For this calculation the fuel building ventilation isolation signal (FBVIS) generated by the channels associated with radiation monitors GGRE0027/0028 will result in a control room ventilation isolation signal (CRVIS). The channels associated with GGRE0027 and GGRE0028, which monitor the fuel building ventilation exhaust, are not response time tested with respect to control room isolation and mitigation of control room doses. However, those channels will be response time tested per new SR 3.3.8.6 with respect to placing the Emergency Exhaust System in the FBVIS mode for the mitigation of offsite radiological consequences.

The licensing basis χ/Q atmospheric dispersion factors are based on extremely calm atmospheric conditions with very low wind velocities. After the radioactivity emerges from the unit vent (the discharge from the emergency exhaust system (EES) fans is directed to the unit vent), it will take a relatively long time to reach the normal control room air intake. For cases that initiate the CRVIS remotely from the control room air intake, this more than offsets any signal processing and damper stroke time considerations. Should different weather conditions be postulated with higher wind

RAI 1.(a) Response (cont.)

speeds, the increased turbulence would provide a significant dilution benefit that would more than offset the earlier arrival of radioactivity at the ventilation intake. This approach is consistent with RIS 2001-19, "*Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests,*" with respect to events that do not rely on the control room intake radiation monitors for control room isolation. Although the FHA dose analyses were not impacted by the replacement SG modification project, Callaway License Amendment 168 (CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING THE STEAM GENERATOR REPLACEMENT PROJECT (TAC NO. MC4437), dated September 29, 2005, Safety Evaluation Section 3.5) acknowledged Callaway's application of this concept as discussed in detail in Section 4.1 of Attachment 1 (pages 15-16) to the RSG amendment application (ULNRC-05056 dated September 17, 2004).

1.(b) The only change to the current radiological consequence analysis of record for a fuel handling accident (FHA) in the fuel building is the 90-second emergency exhaust system (EES) response time.

All parameters, assumptions, and methodologies used in the determination of the radiological consequences of an FHA in the fuel building are documented in Attachment 2 to this RAI response which includes copies of FSAR Section 15.7.4 (with a markup on page 15.7-12 to reflect the 90-second EES response time and the new TS 3.3.8 Surveillance Requirement being added), FSAR Table 15.7-2 (with a markup on sheet 7 to reflect current dose conversion factors endorsed by the NRC since the time RG 1.25 was issued), FSAR Table 15.7-7 (with a markup to reflect the 90-second EES response time), FSAR Table 15.7-8 (with markups to reflect the revised offsite doses for the FHA in the fuel building), FSAR Table 15A-1, FSAR Table 15A-2, FSAR Table 15A-3, FSAR Table 15A-4 (with a markup to reflect current dose conversion factors endorsed by the NRC which are used for the FHA dose analyses), FSAR Table 16.3-2 (with markups to sheets 3 and 5 to reflect the new 90-second EES response time limit), and FSAR Figure 15A-2.

The most recent license amendment which involved an NRC review of the radiological consequences of an FHA in the fuel building was LA129 for the spent fuel pool rerack. It is noted that since LA129 the minimum decay time has been changed from 100 hours to 72 hours (done under an internal 10 CFR 50.59 evaluation in which it was concluded that the resulting dose increases were less than minimal as defined for that rule). In addition, more current NRC-approved dose conversion factors have been used.

The following excerpts are taken from LA129 with emphasis added to the assumptions that have changed since 1999.

<u>RAI 1.(b) Response</u> (cont.)

AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. NPF-30, CALLAWAY PLANT, UNIT 1 (TAC NO. MA1113), January 19, 1999 (corrected on February 4, 1999), Safety Evaluation Section 3.6.3 (emphasis added):

"3.6.3 <u>Design Basis Accidents</u>

In its application, the licensee evaluated the possible consequences of a fuel handling accident (FHA) to determine the thyroid and whole-body doses at the exclusion area boundary (EAB), low population zone (LPZ), and control room. The proposed Callaway SFP reracking will not affect any of the assumptions or inputs used in evaluating the dose consequences of the FHA.

The staff reviewed the licensee's analysis and performed confirmatory calculations to check the acceptability of the licensee's doses. In performing these calculations, the staff used the assumptions of RG 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." For Callaway Plant, Unit 1, an FHA occurring in the reactor building would result in a higher dose to the control room operator than for an FHA in the fuel building. For an FHA in the reactor building, the staff assumed that the cladding of 317 fuel rods (one full assembly plus 20 percent of the rods of an additional assembly) would be ruptured if a fuel assembly were dropped during handling. The damaged fuel rods are assumed to contain freshly off-loaded fuel with a **minimum of 100 hours of decay**. The parameters which the staff utilized in its assessment are presented in Table 1.

The staff's calculations confirmed that the thyroid doses at the EAB, LPZ, and in the control room resulting from a postulated FHA meet the acceptance criteria and that the licensee's calculations are acceptable. The results of the staff's calculations are presented in Table 2. For an FHA, the staff calculated a dose of 62.5 rem thyroid at the EAB and 8.76 rem thyroid at the LPZ. The acceptance criterion at the EAB and LPZ for these accidents is contained in SRP Section 15.7.4 of NUREG-0800 and is 75 rem thyroid dose (25 percent of 10 CFR Part 100 guidelines of 300 rem). In calculating the dose to the control room operator from an FHA, the staff assumed an iodine protection factor (IPF) of 80.3. This is the same IPF that was used in the staff's evaluation of an FHA for an open containment airlock at Callaway Plant, Unit 1 (Callaway Plant, Unit 1 License Amendment No 114, dated July 15, 1996). The staff calculated a dose to the control room operator of 3.94 rem thyroid. The acceptance criterion for the control room operator is 30 rem thyroid (NUREG-0800, SRP Section 6.4). The staff therefore finds the proposed Callaway SFP reracking modification to be acceptable with respect to potential radiological consequences as a result of a postulated fuel handling accident."

RAI 1.(b) Response (cont.)

Table 1

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT AT CALLAWAY PLANT. UNIT 1

Parameters

Power Level, Mwt	3636
Number of Fuel Rods Damaged (1 assembly plus 20%)	317
Total Number of Rods in Core	50,952
Shutdown Time, hours	**100**
Power Peaking Factor	1.65
Fission-Product Release Fractions (%)*	
Iodine (corrected for extended burnup)	12
Noble Gases	30
Pool Decontamination Factors*	
Iodine	100
Noble Gases	1
Iodine Forms (%)*	
Elemental	75
Organic	25
Filter Efficiencies for Control Room (%)	
Elemental	90
Organic	90
Iodine Protection Factor (IPF)	80.3
Atmospheric Dispersion Factors, χ/ Q(sec/m3)	
Exclusion Area Boundary (0-2 hours)**	1.5×10^{-4}
Low Population Zone (0-8 hours)**	2.1×10^{-5}
Control Room (0-8 hours)**	7.6×10^{-4}
Dose Conversion Factors per ICRP 2	

* Regulatory Guide 1.25
** Staff calculated

As previously stated, the "shutdown time," or minimum decay time controlled by FSAR Section 16.9.5, was changed to 72 hours via an Ameren Missouri calculation and internal 10 CFR 50.59 evaluation.

The thyroid and whole body dose conversion factors were approved by the NRC in Callaway LA178, "CALLAWAY PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE: REVISION TO REACTOR COOLANT SYSTEM SPECIFICS ACTIVITY (TAC NO. MD1814)," dated December 18, 2006. The use of ICRP-30 thyroid dose conversion

<u>RAI 1.(b) Response</u> (cont.)

factors was also previously approved in Callaway LA159 with the specific allowance to use them in any future dose calculation as discussed below.

LA159, CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: MAIN FEEDWATER/AUXILIARY FEEDWATER MODIFICATION AND STEAM GENERATOR TUBE RUPTURE RE-ANALYSIS (TAC NOS. MB9875 AND MB9876), March 11, 2004, NRC Safety Evaluation Section 3.2.2.1 (emphasis added):

> "3.2.2.1 Use of ICRP DCFs and Iodine Spike Multiplier of 335
>
> **The licensee's intent in requesting approval of the use of RG 1.195 at Callaway was to gain NRC approval for the use of the ICRP-30 DCFs** and the 335 iodine spiking factor, **on a forward-fitting basis, for all of the Callaway FSAR Chapter 15 accident radiological consequence analyses**. In this regard:
>
> - The NRC staff has previously accepted the use of an iodine spike multiplier of 335 for SGTRs for other licensees. This factor does not apply to any other DBA accident. In reviewing the SGTR with overfill analysis described by the licensee in this amendment request, the NRC staff determined that this value was appropriate for the analysis of SGTR with overfill, and that it would also be acceptable for the other DBA SGTR analyses in the Callaway licensing basis.
>
>   **In RIS 2001-19, the NRC staff identified the thyroid DCFs based on ICRP-30 to be an acceptable change in methodology that does not warrant prior review. This conclusion is consistent with the 10 CFR 50.59 implementation guidance in that the NRC staff has accepted the ICRP-30 DCFs for other licensees and that there is nothing inherently specific to a site, plant, licensee, or accident sequence regarding these factors. As such, the NRC staff concludes that the approval sought by the licensee for use of the ICRP-30 DCFs is granted.**
>
> Therefore, as stated above, the NRC staff concludes that the use of the iodine spike multiplier of 335 for the STGR accident **and the use of ICRP-30 thyroid DCFs at Callaway is acceptable**."

<u>1.(c)</u> The nature of the equipment covered by TS 3.3.8, "EES Actuation Instrumentation," is such that only an FHA in the fuel building is mitigated by this equipment. The fuel building exhaust gaseous radiation channels provide a mitigation function only for events that release radioactivity to the fuel building atmosphere. The only such event in the Callaway FSAR Chapter 15 analysis of DBAs and transients is an

RAI 1.(c) Response (cont.)

FHA in the fuel building. This is discussed in the Applicable Safety Analyses sections of the Bases for TS 3.3.7 and TS 3.3.8 as revised in Attachment 4 of the amendment application. This is also reflected by the LCO 3.3.8 Applicability footnote (a) of TS Table 3.3.8-1 that requires this instrumentation only during movement of irradiated fuel assemblies in the fuel building.

1.(d) As stated in the 3rd paragraph of the amendment application cover letter (ULNRC-05744 dated December 10, 2010), the LPZ and EAB dose changes to FSAR Table 15.7-8 will be included in a future FSAR update per 10 CFR 50.71(e). No changes to Table 15.7-8 will be made with respect to control room doses since control room doses are only reported for the limiting event, a design basis large break LOCA, per Callaway's current licensing basis.

ATTACHMENT 2

FHA ANALYSIS ASSUMPTIONS
AND
ASSOCIATED FSAR MARKUPS

15.7.2.5.3 Conclusions

15.7.2.5.3.1 Filter Loadings

The filter loading due to a liquid radwaste tank rupture does not establish the necessary design margin for the control room intake filters. Thus, the filter loading was not evaluated.

15.7.2.5.3.2 Doses to Receptor at the Exclusion Area Boundary and the Low-Population Zone Outer Boundary

The radiological consequences resulting from the occurrence of a postulated liquid radwaste tank rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2-hour dose at the exclusion area boundary and for the duration of the accident at the low-population zone outer boundary. The results are listed in Table 15.7-6. The resultant dose is well within the guideline values of 10 CFR 100.

15.7.3 POSTULATED RADIOACTIVE RELEASE DUE TO LIQUID TANK FAILURES

This analysis is presented in Section 2.4.13 of the Site Addendum.

15.7.4 FUEL HANDLING ACCIDENTS

The postulated fuel handling accident has been analyzed for two cases: Case 1, a fuel handling accident outside the containment, and Case 2, a fuel handling accident inside the reactor building.

15.7.4.1 Identification of Causes and Accident Description

The accident is defined as the dropping of a spent fuel assembly onto the fuel storage area floor, refueling pool floor, or cask loading pool, resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures.

15.7.4.2 Sequence of Events and Systems Operations

The first step in fuel handling is the safe shutdown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. It is estimated that the earliest time to first fuel transfer after shutdown is 72 hours.

The fuel handling accident is assumed to occur after a fuel assembly has been transferred through the fuel storage pool transfer gate but before it has been placed in its designated location in the fuel storage racks.

15.7.4.3 Core and System Performance

The fuel handling accident in the fuel building does not impact the integrity of the core or its system performance.

15.7.4.4 Barrier Performance

A barrier between the released activity and the environment is the reactor building and the fuel building. Since these buildings are designed seismic Category I, it is safe to assume that during the course of a fuel handling accident their integrity is maintained. This means that the pathway for release of radioactivity for a postulated accident in the fuel building is initially via the auxiliary/fuel building normal exhaust system. After it is isolated on a high radiation signal, the release pathway is via the ESF emergency filtration system. For a postulated accident in the reactor building, the release consists of the total amount of radioactivity which could potentially be released. The fuel storage pool and the refueling pool provide minimum decontamination factors of 100 for iodine.

15.7.4.5 Radiological Consequences

15.7.4.5.1 Method of Analysis

15.7.4.5.1.1 Physical Model

The possibility of a fuel-handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel-handling operations (refer to Section 9.1.4). All refueling operations are conducted in accordance with prescribed procedures.

When transferring irradiated fuel from the core to the fuel storage pool for storage, the reactor cavity and refueling pool are filled with borated water at a boron concentration equal to that in the fuel storage pool, which ensures subcritical conditions in the core even if all rod cluster control (RCC) assemblies were withdrawn. After the reactor head and rod cluster control drive shafts are removed, fuel assemblies are lifted from the core, transferred vertically to the refueling pool, placed horizontally in a conveyor car and pulled through the transfer tube and canal, upended and transferred through the fuel storage pool transfer gate, then lowered into steel racks for storage in the fuel storage pool in a pattern which precludes any possibility of a criticality accident.

The irradiated fuel assemblies may be transferred into the new fuel elevator basket located on the cask loading pit for reconstitution or other fuel repair activities. After the reconstitution or repair is completed, the fuel assemblies are returned to the fuel storage pool storage racks.

Fuel-handling manipulators and hoists are designed so that the fuel cannot be raised above a position that provides an adequate water shield depth for radiation protection of operating personnel.

The containment, fuel building, refueling cavity, refueling pool, and fuel storage pool are designed to seismic Category I requirements, which prevent the structures themselves from failing in the event of a safe shutdown earthquake. The fuel storage racks are also designed to prevent any credible external missile from reaching the stored irradiated fuel. The fuel-handling manipulators, cranes, trollies, bridges, and associated equipment above the water cavities through which the fuel assemblies move are designed to prevent this equipment from generating missiles and damaging the fuel. The construction of the fuel assemblies precludes damage to the fuel should portable or hand tools drop on an assembly.

A fuel-handling accident could occur during the transfer of a fuel assembly from the core to its storage position in the fuel storage pool. Also, such accident could occur during handling of an irradiated fuel assembly associated with reconstitution or other fuel repair activities. The facility is designed so that heavy objects, such as the spent fuel shipping cask, cannot be carried over or tipped over onto the irradiated fuel stored in the fuel storage pool. Only one fuel assembly can be handled at a time. Movement of equipment handling the fuel is kept at low speeds while exercising caution that the fuel assembly does not strike another object or structure during transfer from the core to its storage position. In the unlikely event that an assembly becomes stuck in the transfer tube, natural convection will maintain adequate cooling.

a. Reactor Building Accident

During fuel-handling operations, the containment is kept in an isolatable condition, with the personnel air lock, the emergency air lock, and the containment hatch and penetrations with direct access to the outside atmosphere either closed or capable of being closed. Containment isolation may be initiated either by manual action or on automatic signal from one of the redundant radation monitors, indicating that radioactivity is above the prescribed limits. Personnel air lock doors, emergency air lock doors, and the containment equipment hatch may be open under administrative controls during core alterations or during the movement of irradiated fuel within containment. After an event, the doors are promptly closed. The accident analysis assumes that a direct pathway exists between contaiment and the outside atmosphere for the entire duration of the post-accident release.

In addition to the area radiation monitors in the containment, portable monitors capable of sounding audible alarms are to be located in the fuel-handling area. Should a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors would sound an audible alarm, personnel would be evacuated and the containment would be isolated. The purge and vent lines are closed on a containment isolation signal, thus minimizing the escape of any radioactivity. The containment purge isolation signal may be initiated by manual action.

b. Fuel Building Accident

In the fuel building, a fuel assembly could be dropped in the transfer canal, in the fuel storage pool or in the cask loading pool.

In addition to the area radiation monitors located on the wall around the fuel storage pool, portable radiation monitors capable of emitting audible alarms are located in this area during fuel-handling operations. The doors in the fuel building are closed to maintain controlled leakage characteristics in the fuel storage pool region during operations involving irradiated fuel. Should a fuel assembly be dropped in the canal, in the cask loading pit, or in the pool and release radioactivity above a prescribed level, the radiation monitors sound an audible alarm.

If one of the redundant discharge vent radiation monitors, GG-RE-27 or 28, indicates that the radioactivity in the vent discharge is greater than the prescribed levels, an alarm sounds and the auxiliary/fuel building normal exhaust is switched to the ESF emergency exhaust system to allow the spent fuel pool ventilation to exhaust through the ESF charcoal filters to remove most of the halogens prior to discharging to the atmosphere via the unit vent. The supply ventilation system servicing the spent fuel pool area is automatically shut down, thus ensuring controlled leakage to the atmosphere through charcoal adsorbers (refer to Section 9.4.2).

The probability of a fuel-handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility, as previously mentioned.

15.7.4.5.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.7-7 and 15A-1.

In the evaluation of the fuel-handling accident, all the fission product release assumptions of Regulatory Guide 1.25 have been followed. Table 15.7-2 provides a comparison of the design to the requirements of Regulatory Guide 1.25. The following assumptions, related to the release of fission product gases from the damaged fuel assembly, were used in the analyses:

a. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, for the analyses for the accident in the reactor building the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly.

b. The assembly fission product inventories are based on a radial peaking factor of 1.65.

c. The accident occurs 72 hours after shutdown, which is the earliest time fuel-handling operations can begin. Radioactive decay of the fission product inventories was taken into account during this time period.

d. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation was assumed to be available for immediate release to the water following clad damage.

e. The gap activity released to the fuel pool from the damaged fuel rods consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine contained in the fuel rods at the time of the accident.

f. The pool decontamination factor is 1.0 for noble gases.

g. The effective pool decontamination factor is 100 for iodine.

h. The iodine above the fuel pool is assumed to be composed of 75 percent inorganic and 25 percent organic species.

i. The activity which escapes from the pool is assumed to be available for release to the environment in a time period of 2 hours.

j. No credit for decay or depletion during transit to the site boundary and outer boundary of the low-population zone is assumed.

k. No credit is taken for mixing or holdup in the fuel building atmosphere. The filter efficiency for the ESF emergency filtration system is assumed to be 90 percent for all species of iodine.

l. The fuel building is switched from the auxiliary/fuel building normal exhaust system to the ESF emergency exhaust system within 90 ~~32.1~~ seconds from the time the activity reaches the exhaust duct. The activity released before completion of the switchover is assumed to be discharged directly to the environment with no credit for filtration or dilution. Even if fuel building ventilation isolation does not occur automatically, the calculated doses will be less than those ~~these~~ reported in Table 15.7-8 for the bounding case, inside the reactor building. Response time testing is ~~not~~ required per Technical Specification 3.3.8 for the ~~for any of the~~ fuel building ventilation isolation function~~s~~.

m. For the inside the reactor building case, no credit has been taken for the mixing or holdup of the radioactivity in the reactor building atmosphere. It is assumed that no containment coolers or hydrogen mixing fans are operating.

n. All gap activity assumed available for release is assumed to be released over two hours.

15.7.4.5.1.3 Mathematical Models Used in the Analysis

Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are described in Appendix 15A, Section 15A.2.

b. The atmospheric dispersion factors are calculated, based on the onsite meteorological measurements programs described in Section 2.3 of the Site Addendum, and are provided in Table 15A-2.

c. The thyroid inhalation and total-body immersion doses to a receptor located at the exclusion area boundary and outer boundary of the low population zone are described in Appendix 15A, Sections 15A.2.4 and 15A.2.5, respectively.

15.7.4.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity

For evaluating the radiological consequences due to the postulated fuel-handling accident, the resultant activity is conservatively assumed to be released to the environment during the 0-2-hour period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.4.5.2 Identification of Uncertainties and Conservatisms in Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a fuel-handling accident result from assumptions made involving the amount of fission product gases available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

a. It is assumed in the analysis that all the fuel rods in the dropped assembly are damaged. This is a highly conservative assumption since, transferring fuel under strict fuel handling procedures, only under the worst possible circumstances could the dropping of a spent fuel assembly result in damage to all the fuel rods contained in the assembly.

b. The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. It has been conservatively assumed that the core has been operating at 100 percent for the entire burnup period. The gap activities are listed in Table 15A-3.

c. Iodine removal from the released fission product gas takes place as the gas rises to the pool surface through the body of liquid in the spent fuel pool. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution. The values used in the analysis result in a release of activity approximately a factor of 5 greater than anticipated. The release of activity from the pool to the containment atmosphere is time-dependent and consequently there would be sufficient time for this activity to mix homogeneously in a significant percent of the containment volume.

d. The ESF emergency filtration system charcoal filters are known to operate with at least a 99-percent efficiency. This means a further reduction in the iodine concentrations and thus a reduction in the thyroid doses at the exclusion area boundary and the outer boundary of the low-population zone.

e. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.7.4.5.2.1 Filter Loadings

The ESF filtration systems which function to limit the consequences of a fuel-handling accident in the fuel building are the ESF emergency filtration system and the control room filtration system.

The activity loadings on the control room charcoal adsorbers as a function of time have been evaluated for the loss-of-coolant accident, Section 15.6.5. Since these filters are capable of accommodating the design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated fuel-handling accident releases.

The activity loadings on the ESF filtration system charcoal adsorbers have been evaluated in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg of iodine per gram of activated charcoal.

15.7.4.5.2.2 Doses to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated fuel-handling accident occurring in the fuel building and in the reactor building have been conservatively analyzed, using assumptions and models described in previous sections. The total-body dose due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0-2-hour dose at the exclusion area boundary and for the duration of the accident (0 to 2 hours) at the low-population zone outer boundary. The results are listed in Table 15.7-8. The resultant doses are well within the guideline values of 10 CFR 100.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

The design of the spent fuel cask handling equipment is such that no cask could be dropped more than the equivalent of 30 feet in the air. Therefore, no cask rupture will occur and thus no radioactivity will be released. Refer to Section 9.1.4 for a description of the spent fuel shipping procedures.

TABLE 15.7-2 DESIGN COMPARISON TO THE REGULATORY POSITIONS OF 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" REVISION 0, DATED MARCH 23, 1972

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
1. The assumptions[1] related to the release of radioactive material from the fuel and fuel storage facility as a result of a fuel handling accident are:		
a. The accident occurs at a time after shutdown identified in the technical specifications as the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during the interval between shutdown and commencement of fuel handling operations is taken into consideration.	Complies, except the time after shutdown is identified in Section 16.9.5. Accident occurs 72 hours after shutdown.	Complies, except the time after shutdown is identified in Section 16.9.5. Accident occurs 72 hours after shutdown
b. The maximum fuel rod pressurization[2] is 1200 psig.	Calculations performed as directed by footnote 2 indicate that the assumed pool water decontamination factor is valid for internal pressures up to 1500 psig.	Calculations performed as directed by footnote 2 indicate that the assumed pool water decontamination factor is valid for internal pressures up to 1500 psig.
c. The minimum water depth[2] between the top of the damaged fuel rods and the fuel pool surface is 23 feet.	Complies. Water depth is greater than 23 feet. The release point is assumed to be at the top of the fuel pool storage racks.	Complies. Water depth is greater than 23 feet. The release point is assumed to be at the top of the reactor vessel flange.
d. All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged rods.	Complies.	Complies.
e. The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factors are 1.5 for BWR's and 1.65 for PWR's.	Complies. A peaking factor of 1.65 is used.	Complies. A peaking factor is 1.65 is used.
f. The iodine gap inventory is composed of inorganic species (99.75%) and organic species (.25%).	Complies.	Complies.

TABLE 15.7-2 (Sheet 2)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
g. The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99% of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.	Complies.	Complies.
h. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).	Complies. A decontamination factor of 1 is used.	Complies. A decontamination factor of 1 is used.
i. The radioactive material that escapes from the pool to the building is released from the building[3] over a 2-hour time period.	Complies. A 0-2 hour release from the pool to the building to the environment is assumed.	Complies. A 0-2 hour release from the pool to the building to the environment is assumed.
j. If it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic species and 70% for organic species.[4]	Not applicable; complies with Regulatory Guide 1.52 as described in Table 9.4-2.	No credit is taken for the normal purge filters.
k. The effluent from the filter system passes directly to the emergency exhaust system without mixing[5] in the surrounding building atmosphere and is then released (as an elevated plume for those facilities with stacks[6]).	Complies.	Complies.
2. The assumptions for atmospheric diffusion are: a. Ground Level Releases (1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3)	Short-term atmospheric dispersion factors corresponding to ground level release and accident conditions were based on meteorological measurement programs described in Section 2.3 of the Site Addendum. The dispersion factors are in compliance with the methodology described in Regulatory Guide 1.XXX (see Site Addendum Section 2.3.4.2.1) and represent the worst of the 5 percent overall site meteorology and the 0.5 percent worst sector meteorology.	

TABLE 15.7-2 (Sheet 3)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
Q = amount of material released (curies/sec)		
u = windspeed (meters/sec)		
σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]		
(2) For ground level releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions:		
(a) windspeed of 1 meter/sec;		
(b) uniform wind direction;		
(c) Pasquill diffusion category F.		
(3) Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use of the equation for a ground level release given in regulatory position 2.a.(1) and under the meteorological conditions given in regulatory position 2.a.(2).		

TABLE 15.7-2 (Sheet 4)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(4) Atmospheric diffusion factors for ground level releases may be reduced by a factor ranging from one to a maximum of three (see Figure 2) for additional dispersion produced by the turbulent wake of the reactor building. The volumetric building wake correction as defined in Subdivision 3-3.5.2 of Meteorology and Atomic Energy-1968, is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.		
b. Elevated Releases		
(1) The basic equation for atmospheric diffusion from an elevated release is: $\chi/Q = \frac{e^{-h^2/2\sigma_z^2}}{\pi u \sigma_y \sigma_z}$ Where: χ = the short term average centerline value of the ground level concentration (curies/m^3); Q = amount of material released (curies/sec); u = windspeed (meters/sec); σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]; σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]	Not applicable. Ground level releases were assumed.	Not applicable. Ground level releases were assumed.

TABLE 15.7-2 (Sheet 5)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
h = effective height of release (meters)		
(2) For elevated release; atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions:		
(a) windspeed of 1 meter/sec;		
(b) uniform wind direction;		
(c) envelope of Pasquill diffusion categories for various release heights;		
(d) a fumigation condition exists at the time of the accident.[8]		
(3) Figure 3 is a plot of atmospheric diffusion factors versus distance for an elevated release assuming no fumigation, and Figure 4 is for an elevated release with fumigation.		
(4) Elevated releases are considered to be at a height equal to no more than the actual stack height. Certain site conditions may exist, such as surrounding elevated topography or nearby structures, which will have the effect of reducing the effective stack height. The degree of stack height reduction will be evaluated on an individual case basis.		
3. The following assumptions and equations may be used to obtain conservative approximations of thyroid dose from the inhalation of radioiodine and external whole body dose from radioactive clouds:		
a. The assumptions relative to inhalation thyroid dose approximations are:	Complies. See Appendix 15A, Section 15A.2.4.	Complies. See Appendix 15A, Section 15A.2.4.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		

TABLE 15.7-2 (Sheet 6)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
(2) No correction is made for depletion of the effluent plume of radioiodine due to deposition on the ground, or for the radiological decay of radioiodine in transit.		
(3) Inhalation thyroid doses may be approximated by use of the following equation:		

$$D = \frac{F_g IFPBR(\chi/Q)}{(DF_p)(DF_f)}$$

Where:

D = thyroid dose (rads)

F_g = fraction of fuel rod iodine inventory in fuel rod void space (0.1)

I = core iodine inventory at time of accident (curies)

F = fraction of core damaged so as to release void space iodine

P = fuel peaking factor

B = Breathing rate = 3.47×10^{-4} cubic meters per second (i.e., 10 cubic meters per 8 hour work day as recommended by the ICRP)

DF_p = effective iodine decontamination factor for pool water

DF_f = effective iodine decontamination factor for filters (if present)

χ/Q = atmospheric diffusion factor at receptor location (sec/mμ)

TABLE 15.7-2 (Sheet 7)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
R = adult thyroid dose conversion factor for the iodine isotope of interest (rads per curie). Dose conversion factors for Iodine 131-135 are listed in Table I.[9] These values were derived from "standard man" parameters recommended in ICRP Publication 2.[10]		
TABLE 1 Adult Inhalation Thyroid Dose Conversion Factors Iodine Isotope — Conversion Factor (R) (Rads/curie inhaled) 131 — 1.48×10^6 132 — 5.35×10^4 133 — 4.0×10^5 134 — 2.5×10^4 135 — 1.24×10^5	Table 1; the thyroid dose conversion factors given in ~~Regulatory Guide 1.109~~ ICRP-30 are used. See Table 15A-4.	Table 1; the thyroid dose conversion factors given in ~~Regulatory Guide 1.109~~ ICRP-30 are used. See Table 15A-4.
b. The assumptions relative to external whole body dose approximations are:	Complies. See Appendix 15A, Section 15A.2.5.	Complies. See Appendix 15A, Section 15A.2.5.
(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.		
(2) External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance.	(2) ~~The whole-body dose factors for gammas given in Regulatory Guide 1.109 are used; for iodines, the whole-body dose factors for gammas with credit for 5 cm body tissue attenuation are used.~~ See Table 15A-4 for whole body dose conversion factors from Federal Guidance Report 12.	
For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[11]		

TABLE 15.7-2 (Sheet 8)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
$_{\beta}D'_{\infty} = 0.457\ \bar{E}_{\beta}\chi$ Where: $_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec) $\bar{E}_{\beta}$ = average beta energy per disintegration (Mev/dis) χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3) Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or: $_{\beta}D'_{\infty} = 0.23\ \bar{E}_{\beta}\chi$ For gamma emitting material the dose rate in tissue at the cloud center is: $_{\gamma}D'_{\infty} = 0.507\ \bar{E}_{\gamma}\chi$ Where: $_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec)		

TABLE 15.7-2 (Sheet 9)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
$\bar{E}_\gamma$ = average gamma energy per disintegration (MEV/dis) However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes: $_\gamma D' = 0.25\ \bar{E}_\gamma \chi$ Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as: $_\beta D_\infty = 0.23\ \bar{E}_\beta \psi$ or $_\gamma D = 0.25\ \bar{E}_\gamma \psi$ Where ψ is the concentration time integral for the cloud (curie sec/m^3) (3) The beta and gamma energies emitted per disintegration, as given in Table of Isotopes,[12] are averaged and used according to the methods described in ICRP Publication 2.		

Notes:

1. The assumptions given are valid only for oxide fuels of the types currently in use and in cases where the following conditions are not exceeded:

   a. Peak linear power density of 20.5 kW/ft for the highest power assembly discharged.

   b. Maximum center-line operating fuel temperature less than 4500°F for this assembly.

TABLE 15.7-2 (Sheet 10)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
c. Average burnup for the peak assembly of 25,000 MWD/ton or less (this corresponds to a peak local burnup of about 45,000 MWD/ton).	Gap fractions of 10% remain valid for fuel assemblies up to 33,000 MWD/MTU. Beyond this burnup, a 12% gap fraction will be used.	Gap fractions of 10% remain valid for fuel assemblies up to 33,000 MWD/MTU. Beyond this burnup, a 12% gap fraction will be used.
2. For release pressures greater than 1200 psig and water depths less than 23 feet, the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable to conservatism to those of this guide.		
3. The effectiveness of features provided to reduce the amount of radioactive material available for release to the environment will be evaluated on an individual case basis.		
4. These efficiencies are based upon a 2-inch charcoal bed depth with 1/4 second residence time. Efficiencies may be different for other systems and must be calculated on an individual case basis.		
5. Credit for mixing will be allowed in some cases; the amount of credit will be evaluated on an individual case basis.		
6. Credit for an elevated release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the dispersion of the plume or (b) located far enough from any structure which could affect the dispersion of the plume. For those plants without stacks the atmospheric diffusion factors assuming ground level release given in regulatory position 2.b. should be used.		
7. These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to ensure a conservative estimate of potential offsite exposures.		
8. For sites located more than 2 miles from large bodies of water such as oceans or one of the Great Lakes, a fumigation condition is assumed to exist at the time of the accident and continue for 1/2 hour. For sites located less than 2 miles from large bodies of water a fumigation condition is assumed to exist at the time of accident and continue for the duration of the release (2 hours).		

TABLE 15.7-2 (Sheet 11)

Regulatory Guide 1.25 Position	Case 1 (in Fuel Building)	Case 2 (in Reactor Building)
9. Dose conversion factors taken from "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, J. J. DiNunno, R. E. Baker, F. D. Anderson, and R. L. Waterfield (1962).		
10. Recommendations of the International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation (1959)," ICRP Publication 2, (New York: Permagon Press, 1960).		
11. Meteorology and Atomic Energy-1968, Chapter 7.		
12. C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition (New York: John Wiley and Sons, Inc. 1967).		

TABLE 15.7-7 PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL-HANDLING ACCIDENT

			In Fuel Building	In Reactor Building
I.	Source Data			
	a.	Core power level, MWt	3,636	3,636
	b.	Radial peaking factor	1.65	1.65
	c.	Decay time, hours	72	72
	d.	Number of fuel assemblies affected	1.0	1.2
	e.	Fraction of fission product gases contained in the gap region of the fuel assembly	Per R.G. 1.25	Per R.G. 1.25
II.	Atmospheric Dispersion Factors		See Table 15A-2	See Table 15A-2
III.	Activity Release Data			
	a.	Percent of affected fuel assemblies gap activity released	100	100
	b.	Pool decontamination factors		
		1. Iodine	100	100
		2. Noble gas	1	1
	c.	Filter efficiency, percent	0 until isolation 90 thereafter	0
	d.	Building mixing volumes assumed, percent of total volume	0	0
	e.	HVAC exhaust rate, cfm	20,000 until isolation 9,000 thereafter	Activity completely released over 2 hours.
	f.	Building isolation time,	~~32.1~~ 90 sec	2 hours
	g.	Activity release period, hrs	2	2

TABLE 15.7-8 RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

	Doses (rem)
In Fuel Building	
Exclusion Area Boundary (0-2 hr)	
Thyroid	~~5.59~~ 6.40
Whole-body	~~0.234~~ 0.235
Low Population Zone Outer Boundary (duration)	
Thyroid	~~0.559~~ 0.640
Whole-body	~~0.0234~~ 0.0235
In Reactor Building	
Exclusion Area Boundary (0-2 hr)	
Thyroid	61.7
Whole-body	0.359
Low Population Zone Outer Boundary (duration)	
Thyroid	6.17
Whole-body	0.0359

TABLE 15A-1 PARAMETERS USED IN ACCIDENT ANALYSIS

I.	General			
	1.	Core power level, Mwt		3636 (102% power)
	2.	Number of fuel assemblies in the core		193
	3.	Maximum radial peaking factor		1.65
	4.	Percentage of failed fuel		1.0
	5.	Steam generator tube leak, lb/hr		500
II.	Sources			
	1.	Core inventories, Ci		Table 15A-3
	2.	Gap inventories, Ci		Table 15A-3
	3.	Primary coolant specific activities, μCi/gm		Table 15A-5
	4.	Primary coolant activity, technical specification limit for iodines - I-131 dose equivalent, μCi/gm		1.0
	5.	Secondary coolant activity technical specification limit for iodines - I-131 dose equivalent, μCi/gm		0.1
III.	Activity Release Parameters			
	1.	Free volume of containment, ft^3		2.5×10^6
	2.	Containment leak rate		
		i.	0-24 hours, % per day	0.2
		ii.	after 24 hrs, % per day	0.1
IV.	Control Room Dose Analysis (for LOCA)			
	1.	Control building		
		i.	Mixing volume, cf	150,000
		ii.	Filtered intake, cfm	
			Prior to operator action (0-30 minutes)	900
			After operator action (30 minutes - 720 hours)	450
		iii.	Unfiltered inleakage, cfm	**
		iv.	Filter efficiency (all forms of iodine), %	95
	2.	Control room		
		i.	Volume, cf	100,000
		ii.	Filtered flow from control building, cfm	440

TABLE 15A-1 (Sheet 2)

	iii.	Unfiltered flow from control building, cfm	
		Prior to operator action (0-30 minutes)	440
		After operator action (30 minutes - 720 hours)	0
	iv.	Filtered recirculation, cfm	1360
	v.	Filter efficiency (all forms of iodine), %	95
	vi.	Unfiltered in leakage, cfm	**
V.	Miscellaneous		
	1.	Atmospheric dispersion factors, χ/Q sec/m^3	Table 15A-2
	2.	Dose conversion factors	
	i.	total body and beta skin, rem-meter^3/Ci-sec (Sv-meter^3/Bq-sec)	Table 15A-4
	ii.	thyroid, rem/Ci (Sv/Bq)	Table 15A-4
	3.	Breathing rates, meter^3/sec	
	i.	control room at all times	3.47×10^{-4}
	ii.	offsite	
		0-8 hrs	3.47×10^{-4}
		8-24 hrs	1.75×10^{-4}
		24-720 hrs	2.32×10^{-4}
	4.	Control room occupancy fractions	
		0-24 hrs	1.0
		24-96 hrs	0.6
		96-720 hrs	0.4

** See Figure 15A-2 for inleakage values used in the accident analysis.

TABLE 15A-2 LIMITING SHORT-TERM ATMOSPHERIC DISPERSION FACTORS (χ/QS) FOR ACCIDENT ANALYSIS

Location Type/ Time Interval (hrs)	χ/Q (sec/meter^3)
Site boundary	
0-2	1.5E-4
Low-population zone	
0-8	1.5E-5
8-24	1.0E-5
24-96	4.6E-6
96-720	1.5E-6
Control room (via containment leakage)	
0-8	7.2E-4
8-24	5.3E-4
24-96	1.7E-4
96-720	0
Control room (via unit vent exhaust)	
0-8	1.3E-4
8-24	9.0E-5
24-96	4.1E-5
96-720	0

TABLE 15A-3 FUEL AND ROD GAP INVENTORIES - CORE (CI)

Isotope	Core Fuel	Core Gap
I-131	9.95E+7	9.95E+6
I-132	1.44E+8	1.44E+7
I-133	2.04E+8	2.04E+7
I-134	2.25E+8	2.25E+7
I-135	1.91E+8	1.91E+7
Kr-83m	1.27E+7	1.27E+6
Kr-85m	2.72E+7	2.72E+6
Kr-85	8.61E+5	2.58E+5
Kr-87	5.24E+7	5.24E+6
Kr-88	7.38E+7	7.38E+6
Kr-89	9.03E+7	9.03E+6
Xe-131m	1.12E+6	1.12E+5
Xe-133m	6.35E+6	6.35E+5
Xe-133	1.99E+8	1.99E+7
Xe-135m	3.96E+7	3.96E+6
Xe-135	4.38E+7	4.38E+6
Xe-137	1.78E+8	1.78E+7
Xe-138	1.70E+8	1.70E+7

*Gap activity is assumed to be 10 percent of fuel activity for all isotopes except for Kr-85; for Kr-85 it is assumed to be 30 percent of the fuel activity.

TABLE 15A-4 DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Total Body Rem-meter^3 / Ci-sec	Beta Skin Rem-meter^3 / Ci-sec	Thyroid * Rem/Ci
I-131	8.72E-2	3.17E-2	1.49E+6
I-132	5.13E-1	1.32E-1	1.43E+4
I-133	1.55E-1	7.35E-2	2.69E+5
I-134	5.32E-1	9.23E-2	3.73E+3
I-135	4.21E-1	1.29E-1	5.60E+4
Kr-83m	2.40E-6	0	NA
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Kr-89	5.27E-1	3.20E-1	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-137	4.51E-2	3.87E-1	NA
Xe-138	2.80E-1	1.31E-1	NA

* The radiological consequences for the replacement SG program (1) have been re-analyzed using the following thyroid dose conversion factors from ICRP-30 and whole body dose conversion factors from Federal Guidance Report 12 (except that RG 1.109 Table B-1 is used for Kr-89 and Xe-137). These factors may be applied to other accident sequences as they are re-analyzed (e.g., the fuel handling accident cases addressed in Section 15.7.4):

Nuclide	Total Body **REM-meter^3 / Ci-sec	Thyroid Rem/ci
I-131	6.73E-02	1.07E+06
I-132	4.14E-01	6.29E+03
I-133	1.09E-01	1.81E+05
I-134	4.81E-01	1.07E+03
I-135	2.95E-01	3.14E+04
Kr-83m	5.55E-06	NA
Kr-85m	2.77E-02	NA
Kr-85	4.40E-04	NA

TABLE 15A-4 (Sheet 2)

Nuclide	Total Body **REM-meter^3 / Ci-sec	Thyroid Rem/ci
Kr-87	1.52E-01	NA
Kr-88	3.77E-01	NA
Kr-89	5.27E-01	NA
Xe-131m	1.44E-03	NA
Xe-133m	5.07E-03	NA
Xe-133	5.77E-03	NA
Xe-135m	7.55E-02	NA
Xe-135	4.40E-02	NA
Xe-137	4.51E-02	NA
Xe-138	2.14E-01	NA

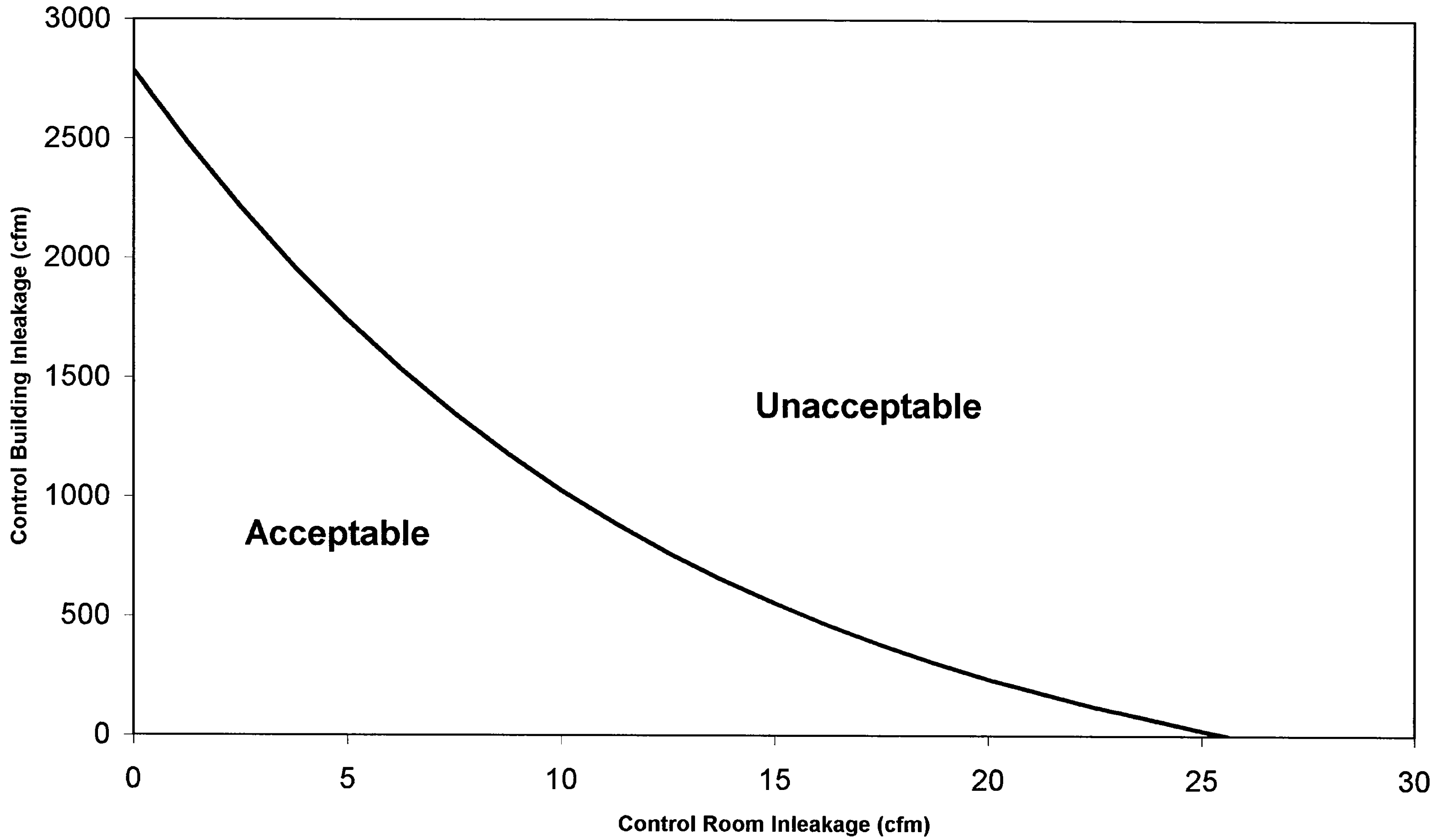
**Federal Guidance Report 12 uses units of $\frac{Sv - meter^3}{Bq - sec}$.

Conversion factors are: 1 Sv = 100 Rem and 1 Bq = 2.7E-11 Ci. The above WB dose conversion factors are equal to those in Federal Guidance Report 12.

(1) FSAR sections re-analyzed for radiological consequences as part of the replacement steam generator program include:

15.1.5 STEAM SYSTEM PIPING FAILURE
15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES
15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)
15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS
15.6.2 BREAK IN INSTUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT
15.6.3 STEAM GENERATOR TUBE FAILURE

Allowable Inleakage Values



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CALLAWAY PLANT
FIGURE 15A-2 ALLOWABLE INLEAKAGE VALUES

TABLE 16.3-2 (Sheet 3)

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
	b.	Start Turbine-Driven Auxiliary Feedwater Pump	$\leq 60^{(8)(17)}$
	c.	Feedwater Isolation	$\leq 2^{(5),(8)}$
10.	Loss-of-Offsite Power		
		Start Turbine-Driven Auxiliary Feedwater Pump	$\leq 60^{(9)}$
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	$\leq 60^{(1)}$
13.	RWST Level-Low-Low Coincident with Safety Injection		
		Automatic Switchover to Containment Sump	$\leq 40^{(10)}$
14.	Loss of Power		
	a.	4 kV Bus Undervoltage-Loss of Voltage	$\leq 14^{(6)}$
	b.	4 kV Bus Undervoltage-Grid Degraded Voltage	$\leq 144^{(13)}$
15.	Phase "A" Isolation		
	a.	Control Room Isolation	N.A.
	b.	Containment Purge Isolation	$\leq 2^{(5)}$
16.	Control Room High Gaseous Activity		
		Control Room Isolation	$\leq 60^{(14)}$

INSERT FSAR 1

TABLE NOTATIONS

(1) Signal actuation, diesel generator starting, and sequencer loading delays included. Valve stroke times and spin-up times for pumps and fans included, as applicable.

(2) Diesel generator starting delay not included. Offsite power available. Signal actuation, sequencer loading, and pump spin-up delays included.

(3) Signal actuation, diesel generator starting and sequencer loading delays included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

INSERT FSAR 1

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
17. Fuel Building Ventilation Exhaust High Gaseous Activity Emergency Exhaust System in the FBVIS Mode	$\leq 90^{(19)}$

TABLE 16.3-2 (Sheet 5)

of non-emergency AC and the loss of normal feedwater accident analyses, initiation of AFW flow is assumed delayed for 90 seconds following reactor trip on a low-low steam generator water level signal.

(17) Response times noted above include the transmitters, 7300 process protection cabinets, solid state protection cabinets, and actuation devices only. For the feedline break accident analysis, initiation of AFW flow is assumed delayed for 90 seconds following reactor trip on low-low steam generator water level signal.

(18) The response time for the reactor trip breakers to open and the gripper release time are satisfied by measurement and included in the response time for each required reactor trip functon.

INSERT FSAR 2

INSERT FSAR 2

(19) 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 place the Emergency Exhaust System in the FBVIS mode of operation.

ATTACHMENT 3

TS 3.3.7 APPLICABLE SAFETY ANALYSES BASES CLARIFICATION

BASES

APPLICABLE SAFETY ANALYSES (continued)

event of a fuel handling accident inside containment. No control room habitablilty mitigation is required for the waste gas decay tank rupture accident. There are no safety analyses that take credit for CREVS actuation upon high containment purge exhaust radiation. A FBVIS is credited to protect the control room in the event of a design basis fuel handling accident (FHA) inside the fuel building.

Sources of control room ventilation isolation signal (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 FHA dose analysis ~~does~~ credits a FBVIS for actuating a CRVIS following a Fuel Handling Accident in the Fuel Building. Due to the remote location of the Fuel Building exhaust 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. INSERT B3.3.7

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

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

(continued)

INSERT B 3.3.7

The channels associated with GGRE0027 and GGRE0028, which monitor the Fuel Building ventilation exhaust, are not response time tested with respect to control room isolation and mitigation of control room doses. However, those channels are response time tested per SR 3.3.8.6 with respect to placing the Emergency Exhaust System in the FBVIS mode for the mitigation of offsite radiological consequences.