15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

15.7.1 MODERATE FREQUENCY INCIDENTS

Incidents in this category are postulated as limiting faults. Moderate frequency incidents will have radiological consequences less severe than the corresponding limiting fault described below.

15.7.2 INFREQUENT INCIDENTS

Incidents in this category are postulated as limiting faults. Infrequent incidents will have radiological consequences less severe than the corresponding limiting fault described below.

15.7.3 LIMITING FAULTS

15.7.3.1 Radioactive Waste Gas System Leak or Failure

→(DRN 04-704, R14)

Deleted.

←(DRN 04-704, R14)

15.7.3.2 Liquid Waste System Leak or Failure (Release to Atmosphere)

→(DRN 04-704, R14)

Deleted.

←(DRN 04-704, R14)

15.7.3.3 Postulated Radioactive Releases Due to Liquid Containing Tank Failures

→(DRN 04-704, R14)
START OF HISTORICAL INFORMATION.

→(DRN 05-1551, R14)
The original waste concentrator of the Waste Management System contained the highest inventory of radionuclides of any component located outside of the Reactor Containment Building. The isotopic inventory was based on a normal operation source term which was based on NUREG-0017. The current source term for core power uprated conditions (3716 WMt) was developed in accordance with ANS 18.1-1999 as discussed in Chapter 11. The analysis discussed below was not revised for power uprate conditions, however is represents a conservative analysis which would be expected to bound plant operations at 3716 MWt.
←(DRN 05-1551, R14)

For purposes of analysis it is conservatively assumed that the entire contents of the waste concentrator tank is instantaneously injected into the Zone 3 aquifer, and is transported as a slug of groundwater without any dispersion or ion-exchange taking place. As indicated in Subsection 2.4.13.3 the maximum groundwater velocity in the site groundwater regime is 0.234 ft/yr in the Zone 3 aquifer. The flow of the groundwater is in a southwest direction away from the Mississippi River. Therefore it would take this contaminated liquid about 1000 years to travel to the nearest boundary (~250 ft.) of the restricted area in this direction. It should be noted that there are no potable water intakes or drinking wells within the restricted area. Due to radioactive decay the total radionuclide concentration at this point after about 1000 years was calculated to be less than 10^{6} of the Maximum Permissible Concentration as defined in Column 2 of Table II of Appendix B to 10CFR20 (Σ C/MPC < $10^{6)}$.
←(DRN 04-704, R14)

→(DRN 00-592; 04-704, R14)
This analysis was based on a liquid waste system which operated on the principle of evaporation. The configuration presently used is based on the principle of demineralization. The demineralizer system was evaluated using very conservative values to determine a maximum saturation activity level (i.e.: system total curies) using FSAR Reactor Coolant Isotopic Inventories to determine a weighted average influent and finally determining a dilution factor representative of the LWM stream. The system spent resin (solid waste) was conservatively treated as a liquid waste.
←(DRN 00-592; 04-704, R14)

Since the groundwater velocity in Zone I is appreciably less than Zone 3, activity level of the nearest surface water is an unrestricted area would be even less.

→(DRN 04-704, R14)
END OF HISTORICAL INFORMATION.
←(DRN 04-704, R14)

15.7.3.4 Design Basis Fuel Handling Accidents

15.7.3.4.1 Identification of Causes and Frequency Classification

The possibility of a fuel handling accident is remote because of the many interlocks and administrative controls and physical limitations imposed on the fuel handling operations (refer to Subsection 9.1.4). All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety and fuel handling.

→(DRN 02-1729, R12-A)
←(DRN 02-1729, R12-A)

Design of the fuel storage racks and handling facilities in both the containment and fuel storage area is such that fuel will always be in a subcritical geometrical array, assuming zero boron concentration in the fuel pool water. The spent fuel pool and refueling pool water contains boron at the refueling boron concentration. Natural convection of the surrounding water provides adequate cooling of fuel during handling and storage. Cooling of the water is provided by the spent fuel pool cooling system. At no time during the transfer from the reactor core to the spent fuel storage rack is a fuel assembly removed from the water. Fuel failure during refueling, as a result of inadvertent criticality or overheating, is not possible.
→(EC-14275, R306)
Interlocks and mechanical stops prevent the spent fuel cask handling crane from moving the cask over stored irradiated fuel in fuel storage racks and limit cask movement. The single-failure-proof main hook on the cask handling crane will move over irradiated fuel in the spent fuel transfer cask, but a fuel handling accident due to a cask drop is not credible since a single-failure-proof handling system is being used.
←(EC-14275, R306)
→(DRN 00-1479, R11-A; 02-1729, R12-A)
←(DRN 00-1479, R11-A; 02-1729, R12-A)

→(DRN 03-179, R12-C; EC-28875 R305)
During fuel handling operations, the containment equipment door, personnel airlock doors, and penetrations may be open provided the equipment door, a minimum of one door in the airlock, and other penetrations are capable of being closed in the event of a fuel handling accident. Each penetration providing direct access to the outside atmosphere shall be either capable of being closed by an isolation valve, blind flange, manual valve or equivalent or capable of being closed on a containment purge isolation signal (CPIS) initiated by redundant area and airborne radiation monitors. Should a fuel handling accident occur inside containment, the equipment door, a minimum of one door in the airlock, and the open penetrations will be closed to minimize the escape of any radioactivity. The containment purge lines are automatically closed upon a CPIS if the fuel handling accident releases activity above prescribed levels.

→(DRN 02-1729, R12-A)

For this evaluation, dropping of a fuel assembly is assumed to occur breaching the cladding and releasing the volatile fission products in the gas gap of the fuel pins. In addition to the area radiation monitor located in the spent fuel cask area, portable radiation monitors capable of emitting audible alarms are located in this area during fuel handling operations. Doors in the Fuel Handling Building are closed to maintain controlled leakage characteristics in the spent fuel pool region during refueling operations involving irradiated fuel. Should a fuel assembly be dropped in the fuel transfer canal or in the spent fuel pool and release radioactivity above a prescribed level, the airborne radiation monitors sound an alarm, alerting personnel to the problem.

←(DRN 02-1729, R12-A)

15.7.3.4.2 Sequence of Events and System Operation

15.7.3.4.2.1 Design Basis Sequence of Events and System Operation

The refueling procedure is described in Subsection 9.1.4. The earliest anticipated time at which a spent fuel assembly could be handled is three days after shutdown.

→(DRN 00-996, R11; EC-5000081470, R301; EC-38571, R307)

For the design basis accident, the failure of two fuel assemblies results in 472 rod failures. The failure of 472 fuel rods is the largest number of fuel rods that could fail from the worst postulated assembly drop.

←(DRN 00-996, R11; EC-5000081470, R301; EC-38571, R307)

15.7.3.4.2.2 Structural Evaluation of Fuel Assembly

In this analysis, dropping of a fuel assembly is assumed. Interlocks and procedural and administrative controls make such an event highly unlikely. However, if an assembly were damaged to the extent that one or more fuel rods were broken, the accumulated fission gases and iodines in the fuel rod gaps would be released to the surrounding water. Release of the solid fission products in the fuel would be negligible because of the low fuel temperature during refueling.

→(DRN 00-996, R11; EC-38571, R307)

The fuel handling accident analysis evaluates the case of a fuel bundle being dropped from the fuel handling device and impacting one fuel bundle located in the spent fuel rack or one or more fuel bundles located in the reactor core. This analysis is performed for the CE 16 NGF and CE 16x16 standard design fuel assemblies that include the weight of components (e.g., CEA and neutron source), handling grapples and some extra weight for margin. The impacted fuel bundles are considered to be struck by either (1) a vertically dropped fuel bundle, or (2) a horizontally dropped fuel bundle. In addition, the analysis considers that the dropped fuel bundle may tip over after impact; the impacted bundle in the reactor core may also tip over after impact. In all cases, energy balance theory is employed to determine the number of damaged fuel rods resulting from the postulated events.

Since the fuel bundle being transported and the impacted bundle(s) are submerged in water, all scenarios take place in water at an assumed temperature of 150°F. As the drop height may vary during movement of a fuel bundle, impact velocities for the vertical and horizontal drop scenarios are conservatively calculated as the terminal velocity of the dropped bundle. In the case of a bundle tipping over, the rotational energy at impact is assumed to be a conservative percentage of the kinetic energy of the horizontally dropped bundle at terminal velocity. The terminal velocity and the resulting impact energy are dependent on many design parameters including: fuel bundle weight and buoyancy, component weight, hydraulic drag, Reynolds number. These factors are all considered in determining the impact velocity and impact energy for each scenario.

In determining the number of fuel rods that fail as a result of each fuel handling accident scenario, all the kinetic energy developed by the dropped, or rotating, fuel assembly is absorbed in the form of fuel rod and guide tube deformation and associated strain energy. Energy absorption by the guide tubes and

←(DRN 00-996, R11; EC-38571, R307)

→(EC-38571, R307)

instrument tube are accounted for by equating each tube is equivalent to 4 fuel rods; this results in the energy being absorbed by 256 fuel rods per assembly. Note that 256 represents the equivalent number of rods for energy absorption; the actual number of rods is 236. This approach is conservative in that it does not take into account the absorption of any impact energy by the upper or lower end fitting components of the dropped and impacted fuel assemblies.

The fuel handling accident analysis also evaluates the case of a 2,000 lb object dropped over the spent fuel rack and impacting one or more fuel assemblies. The terminal velocity of the dropped object and the impact kinetic energy are calculated using the same methodology as that used for dropped fuel bundles. The calculated impact energy is then used to determine the number of failed rods in the impacted fuel bundles.

For the bundle drop scenarios at both the Spent Fuel Pool and Reactor Core locations, the results demonstrate that the maximum total number of fuel rods predicted to fail in the dropped and impacted fuel bundles is 472 rods.

→(DRN 00-996, R11)
←(DRN 00-996, R11)

→(DRN 02-1729, R12-A)
←(DRN 02-1729, R12-A)

←(EC-38571, R307)

15.7.3.4.3 Core and System Performance

This subsection is not applicable for a fuel handling accident.

15.7.3.4.4 Barrier Performance

This subsection is not applicable for a fuel handling accident.

15.7.3.4.5 Radiological Consequences

→(DRN 02-1729, R12-A)

15.7.3.4.5.1 Design Basis Analysis

→(DRN 00-996, R11; 04-704, R14; EC-5000081470, R301; EC-38571, R3-7)

The worst fuel handling accident (assembly drop) that results in failure of 472 fuel rods with an assembly power peaking factor of 1.65 is assumed for the radiological consequences of this event. The assumptions and parameters used in evaluating the fuel handling accident are consistent with Regulatory Guide 1.25 (3/23/72) and Regulatory Guide 1.183 recommendations. All the gap activity in the damaged fuel rods is assumed to be released to the environment within two hours, consistent with the recommendations of RG 1.183, without any filtration. This assumption eliminates the need for filtration of the activity released to the environment from fuel handling building or containment building. The offsite and control room doses are calculated using the RADTRAD computer code and the dose consequences for this event are found to be well below 10CFR50.67 limits.

←(DRN 00-996, R11; 02-1729, R12-A; 04-704, R14; EC-5000081470, R301; EC-38571, R307)

→(DRN 02-1729, R12-A)

←(DRN 02-1729, R12-A)

→(DRN 00-996, R11; 02-1729, R12-A)

←(DRN 00-996, R11; 02-1729, R12-A)

→(DRN 02-1729, R12-A)
Input and Assumptions:

→(DRN 04-704, R14)
The evaluation for the offsite and control room radiological consequences of a FHA used the RADTRAD computer code. The analysis incorporated the appropriate conservative assumptions provided in RG 1.25 and RG 1.183. The following provides the conservative input and assumptions used in the calculations:
←(DRN 04-704, R14)
→(EC-38571, R307)
Number of Failed Fuel Rods: 472 fuel rods in the dropped and impacted bundles are assumed to fail based on the worst postulated assembly drop (Design Basis Accident).
←(EC-38571, R307)
→(DRN 04-704, R14)
Gap Inventory: For each isotope the most conservative calculated activity (i.e., the highest value) for the 30, 40, 50, 60, and 70 GWD/MTU burnups is assumed.

Power Level: 3735 MWt = 3716 * 1.005
←(DRN 04-704, R14)
→(EC-5000081470, R301; EC-38571, R307)
Release Duration: It is assumed that all the activity released from the failed bundles is released to the environment within two hours after the accident.
←(EC-5000081470, R301; EC-38571, R307)
Radiological source term:
The radiological source term used in the fuel handling accident is provided in Table 15.7-6.
←(DRN 02-1729, R12-A)
→(DRN 04-704, R14)
The release fractions are based on NUREG/CR-5009.

Decontamination Factor:

RG 1.25 states that, "For release pressure greater than 1200 psig, the iodine decontamination factor will be less than those assumed in this guide." For Waterford it was determined that the fuel rod pressure exceeds 1200 psig at 49 GWD/MTU at a spent fuel pool temperature of 177°F. However, it was shown that the radiological source term used in Waterford FHA analysis bounds this conditions.
←(DRN 04-704, R14)
→(EC-5000081470, R301)
A minimum control room volume is assumed. FHA Main Control Room dose has negligible sensitivity to control room volume assumptions.
←(EC-5000081470, R301)

15.7.3.4.6 Results

a) Offsite Doses
→(DRN 02-1729, R12-A; 04-704, R14; EC-38571, R307)
The potential radiological consequences resulting from the occurrence of a postulated fuel handling accident have been conservatively analyzed, using inputs, assumptions provided in Table 15.7-6 and models described in the preceding subsections. The TEDE dose at exclusion area boundary (EAB) and low population zone (LPZ) is provided in Table 15.7-7.
←(DRN 02-1729, R12-A; 04-704, R14; EC-38571, R307)

→(DRN 02-1729, R12-A)

b) Dose to Main Control Room Personnel

→(DRN 04-704, R14)

The radiological consequences TEDE doses to main control room personnel following a fuel handling accident are provided in Table 15.7-7. The control room doses are well within the acceptable limits.

←(DRN 02-1729, R12-A; 04-704, R14)

15.7.3.5 Spent Fuel Cask Drop Accidents

→(DRN 00-996, R11; EC-14275, R306)

15.7.3.5.1 Cask Drop Into Spent Fuel Pool, Cask Pit Storage Area, Cask Decontamination Area, and Rail Bay

→(EC-30504, R305)

As discussed in Subsection 9.1.4, the cask handling crane carrying a spent fuel transfer cask or other heavy load (i.e., loads in excess of the weight of a fuel assembly plus handling tool) is prohibited from traveling over the spent fuel pool. Only the single-failure-proof main hook on the cask handling crane is permitted to carry a spent fuel transfer cask or other heavy load over the cask storage area or any unprotected safety related equipment. Thus, an accident resulting from a drop of a spent fuel transfer cask or other heavy load into the spent fuel pool, the cask storage area, the ask decontamination area, or the rail bay is not credible.

←(DRN 00-996, R11; EC-30504, R305)

15.7.3.5.2 Cask Drop to Flat Surface

As discussed in Subsection 9.1.4, loaded spent transfer fuel casks are only handled using the single-failure-proof main hook on the cask handling crane as part of a single-failure-proof handling system, and thus a drop of a spent fuel transfer cask is not credible and is not postulated. Since a drop of a spent fuel transfer cask is not postulated, the radiological consequences of a cask drop accident are not evaluated.

←(EC-14275, R306)

SECTION 15.7: REFERENCES

1. Love, A. E. H., A Treatise on the Mathematical Theory of Elasticity, 4th Edition, Dover Publications, New York, New York, October 1926.

2. Gabrielson, V. K., SHOCK - A Computer Code for Solving Lumped Mass Dynamic Systems, SCL-DR- 5-35, January 1966.

TABLE 15.7-1 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

TABLE 15.7-2 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

TABLE 15.7-3 Revision 14 (12/05)

→(DRN 04-704, R14)

TABLE INTENTIONALLY DELETED.

←(DRN 04-704, R14)

→(DRN 04-704, R14)
START OF HISTORICAL INFORMATION
←(DRN 04-704, R14)

TABLE 15.7 4 Revision 14 (12/05)

MAXIMUM ISOTOPIC INVENTORIES IN EQUIPMENT CLASSIFIED AS
NON SAFETY, NON SEISMIC (CURIES)
WASTE MANAGEMENT SYSTEM

→(DRN 01-1250, R11-B)

Isotope	Waste Tanks A&B (each)	Waste Tank C	Laundry Tank (each)	Waste Cond. Tank (each)	Waste Demineralizer	Waste Cond. Ion Exchanger	Total For System
I 131	5.7(0)	5.7(1)	7.1(3)	4.9(3)	2.8(+1)	4.3(2)	9.7(+1)
I 132	1.6(0)	1.6(1)	2.0(3)	1.4(3)	8.0(0)	1.4(4)	2.7(1)
I 133	7.1(0)	7.1(1)	8.9(3)	6.1(3)	3.6(+1)	5.8(3)	1.2(2)
I 134	6.9(1)	6.9(0)	8.7(4)	6.0(4)	3.5(0)	2.4(5)	1.2(1)
I 135	3.1(0)	3.1(1)	3.9(3)	2.7(3)	1.6(+1)	8.1(4)	5.3(1)

←(DRN 01-1250, R11-B)

BORON MANAGEMENT SYSTEM

Isotope	Pre Concentrator Ion Exchanger (both)	Boric Acid Condensate Ion Exchanger (both)	Boric Acid Concentrator
Kr 85m	0.0	0.0	9.1(2)
Kr 85	0.0	0.0	9.1(0)
Kr 87	0.0	0.0	1.7(2)
Kr 88	0.0	0.0	1.1(1)
I 131	4.0(+1)	1.9(2)	7.6(0)
I 132	2.6(3)	1.3(6)	4.1(2)
I 133	7.2(1)	3.5(4)	1.3(0)
I 134	1.8(4)	8.9(8)	7.7(3)
1 135	3.7(2)	1.8(5)	2.0(1)
Xe 131m	0.0	0.0	2.7(0)
Xe 133	0.0	0.0	2.4(+2)
Xe 135m	0.0	0.0	3.4(3)
Xe 135	0.0	0.0	6.6(1)
Xe 138	0.0	0.0	1.6(3)

→(DRN 04-704, R14)
END OF HISTORICAL INFORMATION
←(DRN 04-704, R14)

→(DRN 04-704, R14)
START OF HISTORICAL INFORMATION
←(DRN 04-704, R14)

TABLE 15.7-5 Revision 14 (12/05)

RADIOLOGICAL EXPOSURES AS A RESULT OF LIQUID WASTE SYSTEM FAILURE

(CURIES)

Isotope	Boron Management System	Waste Management System
Kr-85m	9.1(-2)	0.0
Kr-85	9.1(0)	0.0
Kr-87	1.7(-2)	0.0
Kr-88	1.1(-1)	0.0
I-131	4.8(-1)	9.7(-1)
I-132	4.4(-4)	2.7(-1)
I-133	2.0(-2)	1.2(0)
I-134	7.9(-5)	1.2(-1)
1-135	2.4(-3)	5.3(-1)
Xe-131m	2.7(0)	0.0
Xe-133	2.4(+2)	0.0
Xe-135m	3.4(-3)	0.0
Xe-135	6.6(-1)	0.0
Xe-138	1.6(-3)	0.0

Dose (rem)	Boron Management System	Waste Management System
EAB		
Thyroid	1.6 (-1)	1.6 (-1)
Whole body	1.2 (-3)	2.9 (-4)
LPZ		
Thyroid	1.8 (-2)	1.8 (-2)
Whole body	1.3 (-4)	3.2 (-5)

→(DRN 04-704, R14)
END OF HISTORICAL INFORMATION
←(DRN 04-704, R14)

TABLE 15.7-6 (Sheet 1 of 2) Revision 307 (07/13)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

→(DRN 04-704, R14)

Core Power Level:	3735 MWt
Core Inventory:	(Ci/MWt)
Kr-85	4.520E+02
Kr-85m	1.378E-01
Kr-87	5.849E-14
Kr-88	2.071E-04
I-131	9.867E+01
I-132	3.985E+01
I-133	9.764E+00
I-134	9.488E-23
I-135	4.991E-02
Xe-131m	1.551E+02
Xe-133	1.667E+04
Xe-133m	3.713E+02
Xe-135	2.223E+02
Xe-135m	1.625E+00
Fission Product Gap Fractions:	
I-131	12%
Kr-85	14%
Other Noble Gases	5%
Other Halogens	5%
Alkali Metals	12%

→(EC-38571, R307)

Fuel Rods Failing (maximum):	472 rods

←(EC-38571, R307)

Iodine Chemical Form *:	
Elemental	4.85%
Organic	0.15%
Particulate	95%

* The releases from the pool are conservatively modeled as 99.85% elemental and 0.15% organic iodine.

Control Room Parameters:

→(EC-5000081470, R301)

Volume	168,500 ft^3

←(EC-5000081470, R301)

Recirculation Flow Rate	3800 CFM
Iodine Filter Efficiency	99% (elemental/particulate/organic)
Pressurization Flow	225 CFM

→(EC-5000081470, R301)

Unfiltered Inleakage	100 CFM

←(EC-5000081470, R301)

Breathing Rate	3.47E-04 m^3/sec.
Control Room Occupancy Factors	
0-24 hours	1.0

←(DRN 04-704, R14)

TABLE 15.7-6 (Sheet 2 of 2) Revision 301 (09/07)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

→(DRN 04-704, R14; EC-5000081470, R301)

Main Control Room χ/Q Assumed:

Time	Unfiltered Inleakage East MCR Intake	Pressurization Flow West MCR Intake
0-2 hr	2.77E-03	3.90E-04 *

* Factor of 4 reduction credited per SRP 6.4

←(DRN 04-704, R14; EC-5000081470, R301)

TABLE 15.7-7 Revision 14 (12/05)

→(DRN 02-1729, R12-A)

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT

←(DRN 02-1729, R12-A)

→(DRN 00-996, R11; 02-1729, R12-A; 04-704, R14)

	FHA	**Acceptance Criteria**
EAB (worst two hour dose)	< 6.3	6.3 Rem TEDE
LPZ (duration)	< 6.3	6.3 Rem TEDE
MCR	< 5	5 Rem TEDE

←(DRN 00-996, R11; 02-1729, R12-A; 04-704, R14)

TABLE 15.7-8

INFORMATION NEEDED TO EVALUATE CONTAINMENT ISOLATION CAPABILITY DURING REFUELING ACCIDENT

1.

2.

3.

Security-Related Information
Text Withheld Under 10 CFR 2.390

4.

5.

6.