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RADIOACTIVE GASEOUS EFFLUENT SOURCE TERMS FOR POSTULATED ACCIDENT CONDITIONS IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

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ABSTRACT

Calculations were made to determine radioactive gaseous effluent releases to the atmosphere and to identifiable points within the containment building and other confined spaces resulting from several potential accidents in light-water-cooled nuclear power plants. These "source terms" are intended to provide bases for operating range specification (accident response sensitivity) for radioactive effluent monitoring instrumentation. As expected, the primary contributors to airborne radioactivity resulting from operating reactor accidents are the radioactive noble gases (krypton and xenon) and the halogens (bromine and iodine). Their relative contributions at any given time are dependent upon the time following release from the reactor core fuel element(s).

SUMMARY

This task provides tabular presentations of "source terms" relative to radioactive airborne releases resulting from postulated nuclear power plant accidents. No conclusions are drawn and no recommendations are made.

PREFACE

The primary objective of this task was the calculation of radioactive gaseous effluent releases, also referred to as "source terms," which can be postulated to occur as the result of a spectrum of potential accidents in light-water-cooled nuclear power plants. The calculated source terms will be utilized, together with other data, as bases for the specification of operating ranges for the accident response sensitivity of radioactive effluent monitoring instrumentation.

The accidents considered were categorized according to the "Class 1 to Class 9" designations of Appendix I of NUREG-0099[4] and include eight accident types in Classes 3 through 8 for BWRs and ten accident types in Classes 3 through 8 for PWRs. Both a conservative (termed "maximum") and a "realistic" evaluation were made for most of the accident types.

The analyses differ in certain respects from the analyses which are performed for the "design basis accidents" as described in Section 15 of Regulatory Guide 1.70 and as provided in the applicants' safety analysis reports (SAR). Therefore, the analytical procedures and results in this report should not be construed as meeting the guidelines of Regulatory Guide 1.70 and should not be used as models for design basis analyses.

The assumptions and conditions used in the analyses for "maximum" accidents were based principally on the parameters provided for the evaluation of the environmental effects of accidents in Appendix I of NUREG-0099[4], and on plant-specific parameters for plants representative of the largest plants currently under review by NRC. Parameters for BWRs were those for BWR/6 plants using the GESSAR-251 nuclear steam supply system. Parameters for PWRs were taken from representative SARs for Westinghouse and Combustion Engineering U-tube steam generator plants and for Babcock and Wilcox once-through steam generator plants. Where necessary, parameters were normalized to a reactor thermal power level of 3800 MWt.

The assumptions and conditions for "realistic" accidents (see Regulatory Guide 1.70, Rev. 1, p. 15-7) were generally those of NUREG-0016[2] for BWRs and NUREG-0017[5] for PWRs.

In many instances, where needed accident conditions or assumptions were unavailable from the referenced reports, assumptions were those of the author(s), in consultation with staff members of the U.S. Nuclear Regulatory Commission.

In those cases where the postulated accidents lead to release of gaseous radioactive materials into sealed containments or other confined spaces, numerical values for th peak radioactivity concentrations (by radionuclide) were provided. No attempt was made, in this study, to determine concentrations as a function of time or of the use of internal atmospheric cleanup systems. However, where radioactivity was presumed to be released to the containment atmosphere and subsequently released to the environment prior to containment isolation, the effects of typical on-line effluent reduction features were taken into account. In those cases where releases were directly to the environment, total release quantities and individual radionuclide contributors were calculated.

In all accident cases which involved fuel damage, the ORIGEN (ORNL Isotope Generation and Depletion) code was used to calculate core radionuclide inventories. Inventories of radionuclides in the primary and secondary systems were obtained from NUREG-0016 (BWR)[2] and NUREG-0017 (PWR)[5]. In all cases, a power level of 3800 MWt, for a sufficient time period to establish radionuclide equilibrium in both the primary and secondary coolant systems (where applicable), was used.

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PART I

BOILING WATER REACTORS

I. ACCIDENT CLASS 3.0 – RADWASTE SYSTEM FAILURE

Accident 3.2 – Release of Waste Gas Storage Tank Contents

1. INTRODUCTION

Gases released from the reactor core are either absorbed in the coolant or are released into the steam. It was assumed that all of the noble gases and a small fraction of the halogens go with the steam. The gases that are transported with the steam tend to accumulate in the main condenser because they are noncondensible. Small quantities of gases escape from the steam system through leaks at such points as valve stem packings and turbine bearings. A small amount of noncondensible air inleakage also occurs in the main condenser. The bulk of the noncondensible gases, including the radioactive gases, accumulate in the main condenser where they must be removed before they accumulate in sufficient quantities to interfere with normal condenser operation.

During reactor operation, the noncondensibles in the main condenser are typically removed by two series-connected steam jet air ejectors. During startup and periods of reactor shutdown, condenser vacuum is achieved or maintained by mechanical vacuum pumps.

The noncondensible gases consist of radioactive noble gases, radioiodines, decay products of the radioactive noble gases, radiolytically dissociated hydrogen and oxygen, and air. The gases are sufficiently radioactive to require treatment prior to release and also require treatment to control the potential for hydrogen explosions in the venting system; both of these functions are incorporated in the main condenser off-gas treatment system, which typically includes the air ejectors, preheater, recombiner, moisture condenser, water separator, holdup pipe, cooler condenser, moisture separator, prefilter, dryer, charcoal filled delay tanks, and a posttreatment particulate filter.

Holdup time is dependent on the rate of gas flow through the gas treatment system, on the temperature and mass of charcoal, and on the rate of air inleakage to the condenser.

A significant amount of radioactive gas is accumulated in the operation of the off-gas treatment system and would be released in the case of a gross failure of the delay tanks or plumbing.

2. POSTULATED CAUSES

The equipment and piping are designed to withstand any explosion which has a probability of occurring. Therefore, an explosion is not considered a possible gross failure mode. It is possible for a seal to fail from an explosive reaction, and this failure mode is postulated for the realistic accident case. The most credible event which could cause a gross failure in the system, releasing significant radioactivity to the environment, is an earthquake[1].

3. ACCIDENT DESCRIPTION

Table I shows the assumed gas residence times for the main condenser off-gas treatment system. The longest holdup times are in the charcoal-filled delay tanks located in the equipment vault. The holdup pipe has the longest residence time of the components outside the vault. The points of failure considered are (a) the charcoal delay tanks (for the maximum release case) and (b) the holdup pipe (for the realistic case). It should be noted that Reference 1 conclusions are based on refrigerated charcoal-filled delay tanks, but this study uses ambient-temperature delay tanks (77°F).

TABLE I[a]

GAS RESIDENCE TIMES FOR OFF-GAS TREATMENT SYSTEM

	Time (hr)	
Component	Kr	Xe
Holdup pipe	0.17	0.17
First charcoal-filled delay tank	1.3	23

[a] This evaluation assumes the GE N64 design for the main condenser off-gas treatment system. The N64 design utilizes eight delay tanks in series, with each tank containing three tons of charcoal. While the N64 design features refrigerated delay tanks, this study assumes operation at an ambient temperature of 77°F and a dew point of 0°F.

3.1 Case 1 – Maximum Release Case

A seismic event more severe than the one for which the system is designed is assumed to occur, causing significant damage to the plant structure. The concrete structure surrounding the charcoal-filled delay tank fails, and a circumferential failure of one tank is assumed to occur due to falling concrete. The contents of the first delay tank are assumed to be instantaneously released to the environment. It is assumed that the charcoal train collected 99% of the halogens in the first charcoal delay tank prior to the release.

3.2 Case 2 – Realistic Case

A hydrogen explosion is assumed to occur due to a malfunction of the off-gas recombiner. A seal failure is assumed that permits the contents of the off-gas system to leak to the turbine building from the equipment vault.

4. EFFLUENT RELEASE PATHWAY

4.1 Case 1 – Maximum Release Case

4.1.1 Point of Maximum Concentration. The failure is postulated to have been caused by falling concrete, so it can also be postulated that the vault fails. All of the charcoal is assumed to be displaced from the failed delay tank. The plant will be shut down by the accident, and there will be no additional gas introduced into the main condenser off-gas system. The vault will be the region of maximum concentration. The concentration may be measured at any point in the equipment vault; however, there is a high probability that any instrumentation will be rendered inoperative by the assumed seismic event since the instrumentation is not designed to withstand the event.

4.1.2 Principal Point of Environmental Release. The off-gas system for a typical BWR/6 is located in the turbine building. Since the seismic event is assumed to cause structural failure of the building, the entire amount of activity released is assumed to escape directly to the environment at ground level.

4.2 Case 2 – Realistic Case

4.2.1 Point of Maximum Concentration. The rate of release is insufficient to raise the pressure in the turbine building and is well within the exhaust capability of the heating and ventilating system in the vicinity of the failure. Consequently, the release is confined mainly to the area of the failure. Equilibrium concentrations will occur within minutes of the failure. The point of maximum concentration is taken as the exhaust duct intake nearest the failure point.

4.2.2 Principal Point of Environmental Release. The effluent will be drawn into one of the turbine building exhaust systems where it will be diluted and then released at roof level through an unfiltered vent.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

The maximum values in Reference 2 are used to establish the releases for this analysis. The reactor design power is assumed to be 3800 MWt. Other assumptions relating to the release of radioactive gases from the postulated failures are as follows:

(1) The reactor is operating at design power (3800 MWt) at the time of the accident.

(2) Condenser air inleakage is assumed to be 30 cfm[1].

(3) Off-gas holdup time is 10 minutes (to reach the charcoal tanks)[1]. Xenon residence time in the charcoal tanks is 23 hours[2].

(4) Core fission product inventories were computed by use of the ORIGEN computer code[3].

5.1 Case 1 – Maximum Release Case

(1) The accident results in a tear around the circumference of the first delay tank.

(2) All of the charcoal is displaced from the failed delay tank (3 tons)[1].

(3) Ten percent of the iodine activity in the failed tank is released to the atmosphere.

(4) Two-thirds of the noble gas activity in the delay tank is released to the atmosphere.

(5) Release rate from the fuel to the steam is assumed to be 380,00 μCi/sec (measured at 30-minute decay) for 30 days prior to the accident.

The radioactivity released to the atmosphere, by radionuclide, is shown in Table II.

TABLE II

RADIOACTIVITY RELEASED FROM RUPTURED CHARCOAL DELAY TANK (Maximum Case)

Radionuclide	Total Radioactivity in First Charcoal Tank (Ci)	Radioactivity Released to Atmosphere (Ci)
Kr-83m	5.2(1)[a]	3.5(1)
Kr-85m	1.1(2)	7.1(1)
Kr-85	3.8(-1)	2.5(-1)
Kr-87	2.7(2)	1.8(2)
Kr-88	3.4(2)	2.3(2)
Kr-89	1.8(1)	1.2(1)
Kr-90	2.0(-4)	1.3(-4)
Xe-131m	5.1	3.4
Xe-133m	8.8(1)	5.8(1)
Xe-133	2.0(3)	1.4(3)
Xe-135m	9.9(1)	6.6(1)
Xe-135	3.8(3)	2.5(3)
Xe-137	3.4(1)	2.3(1)
Xe-138	2.9(2)	1.9(2)
Br-83	9.6(-1)	9.6(-3)
Br-84	3.0(-1)	3.0(-3)
Br-85	1.3(-2)	1.3(-4)
I-131	1.3(2)	1.3
I-132	9.1	9.1(-2)
I-133	5.8[illegible]	5.8(-1)
I-134	5.[illegible]	5.3(-2)
I-135	1.8(1)	1.8(-1)
Total	7.3(3)	4.8(3)

[a] Example: $5.2(+1) = 5.2 \times 10^1$.

5.2 Case 2 – Realistic Case

(1) An 0.8 in.^{2} failure occurs in the system near the holdup pipe.

(2) The pressure decays from its normal operating value of 16 psia[1].

(3) The failure is not detected before the ambient concentrations are stabilized.

(4) Based on these assumptions, the outflow rate based on flow through a circular orifice is:

Time (sec)	0	10	20	30	40	50	60	120
Flow Rate (lb/sec)	0.12	0.084	0.052	0.035	0.030	0.029	0.029	0.029

(5) The break is assumed to occur in a turbine building room of 20,000 ft^3 volume which experiences three air changes per hour.

(6) The noble gas release rate to the steam prior to the accident is assumed to be 60,000 μCi/sec, measured at a 30-minute decay[2].

The radioactivity released to the turbine building, as a function of time, is listed in Table III.

TABLE III

RADIOACTIVITY CONCENTRATIONS RESULTING FROM A HOLDUP PIPE BREAK (0.8 in.^{2})
(Realistic Case)

Radionuclide	Total Radioactivity in Turbine Building Room (Ci)			Peak Concentration in Turbine Building (μCi/cc)		
	10 min	1 hr	2 hr	10 min	1 hr	2 hr
Kr-83m	8.2(-1)[a]	1.8	1.9	1.4(-3)	3.2(-3)	3.4(-3)
Kr-85m	1.4	3.4	3.5	2.5(-3)	5.9(-3)	6.2(-3)
Kr-85	4.6(-3)	1.1(-2)	1.2(-2)	8.1(-6)	2.0(-5)	2.1(-5)
Kr-87	4.8	1.1(1)	1.1(1)	8.5(-3)	1.9(-2)	1.9(-5)
Kr-88	5.0	1.1(1)	1.2(1)	8.8(-3)	2.0(-2)	2.1(-2)
Kr-89	1.2(1)	1.3(1)	1.3(1)	2.2(-2)	2.4(-2)	2.4(-2)
Kr-90	3.6	3.6	3.6	6.3(-3)	6.3(-3)	6.3(-3)
Xe-131m	3.6(-3)	8.7(-3)	9.1(-3)	6.4(-6)	1.5(-5)	1.6(-5)
Xe-133m	6.9(-2)	1.7(-1)	1.7(-1)	1.2(-4)	2.9(-4)	3.1(-4)
Xe-133	2.0	4.8	5.0	3.5(-3)	8.4(-3)	8.8(-3)
Xe-135m	5.2	8.4	8.5	9.2(-3)	1.5(-2)	1.5(-2)
Xe-135	5.5	1.3(1)	1.4(1)	9.7(-3)	2.3(-2)	2.4(-2)
Xe-137	1.6(1)	1.8(1)	1.8(1)	2.9(-2)	3.2(-2)	3.2(-2)
Xe-138	1.7(1)	2.7(1)	2.7(1)	3.0(-2)	4.8(-2)	4.8(-2)
Br-83	4.5(-2)	1.0(-1)	1.1(-1)	7.9(-5)	1.8(-4)	1.9(-4)
Br-84	6.9(-2)	1.3(-1)	1.3(-1)	1.2(-4)	2.3(-4)	2.4(-4)
Br-85	2.7(-2)	3.3(-2)	3.3(-2)	4.8(-5)	5.9(-5)	5.9(-5)
I-131	7.7(-2)	1.8(-1)	1.9(-1)	1.4(-4)	3.3(-4)	3.4(-4)
I-132	4.5(-1)	1.0	1.1	7.9(-4)	1.8(-3)	1.9(-3)
I-133	3.1(-1)	7.3(-1)	7.7(-1)	5.4(-4)	1.3(-3)	1.4(-3)
I-134	7.2(-1)	1.5	1.5	1.3(-3)	2.6(-3)	2.7(-3)
I-135	3.0(-1)	7.2(-1)	7.5(-1)	5.4(-4)	1.3(-3)	1.3(-3)
Total	7.5(1)	1.2(2)	1.2(2)	1.3(-1)	2.1(-1)	2.1(-1)

[a] Example: 8.2(-1) = 8.2 x 10^{-1}.

II. ACCIDENT CLASS 4.0 – FISSION PRODUCTS TO PRIMARY SYSTEM (BWR)

Accident 4.2 – Off-Design Transients That Induce Fuel Failure Above Those Expected

1. INTRODUCTION

Fuel failures can be caused by local reductions in reactor core coolant flow resulting from blockage, rod bowing, or blistering. It is also conceivable that the neutron flux can produce abnormal localized heating peaks that can cause local failures. It is postulated that 0.02% of the core inventory of noble gases and halogens are released to the reactor coolant in such an accident[4].

2. POSTULATED CAUSES

The possibility of a gross fuel failure accident is considered to be remote because of the extensive system of design controls, monitors, screens and filters, as well as administrative controls. Whenever the reactor coolant system is opened up, inspections and procedures are used to prevent the introduction of foreign objects that could cause blockage. Rod withdrawal schedules and fuels are arranged to provide optimum flux distributions. Quality control programs are maintained to minimize the occurrence of faulty fuel in the reactor.

3. ACCIDENT DESCRIPTION

A single accident is considered to be representative of this class of accident. A foreign object is assumed to block the coolant flow to 6 fuel bundles in a typical boiling water reactor (BWR). This flow blockage causes departure from nucleate boiling (DNB) to occur and results in cladding failure of all the fuel rods in those bundles (63 rods/bundle x 6 bundles = 378 failed rods). The radioactive gases released are detected by the radiation monitors which initiate a plant protection system (PPS) scram (in time to prevent a meltdown of any of the damaged rods) and isolate the steam system.

The main steamline isolation valves are assumed to receive an automatic closure signal 0.5 second after the released activity reaches the main steamline radiation monitors, and to be fully closed 5 seconds after receipt of the closure signal. Consequently, only the gases in the fuel rod plenum are released to the reactor coolant system. All of the noble gases and 1% of the halogens from the fuel rod plenum are assumed to enter the steam system. The reactor is assumed to be operating at 3800 MWt power at the time of the accident.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. Noble gases constitute most of the release and are assumed to be released to the steam flow at the time of the accident. In the 5.5 seconds from the time of the accident until the steam valve is fully closed, all of the noble gases and 1% of the halogens released to the reactor coolant are assumed to be transported to the main condenser. It was assumed that it takes approximately 8 hours after steam valve closure for the main condenser to reach ambient pressure. At that time, the main condenser begins to discharge its contents to the turbine building (main condenser room) below the operating floor.

4.1.2 Principal Point of Environmental Release. The effluent will be drawn into the exhaust ducts of the heating and ventilating system which carry it to the roof of the turbine building where it is released to the environment.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

Assumptions and conditions related to the release of radioactive gases from the postulated flow blockage accident are as follows:

(1) The reactor had operated at design basis power (3800 MWt) for 1000 days at the time of the accident.

(2) The released gases are only those in the fuel rod plenum at the time of the accident. No fuel melting occurs.

(3) A one-minute decay time is assumed in determining the fission product inventory in the fuel rod plenum. This delay is to account for the fission product migration time[1].

(4) All the fuel rods in the six blocked bundles are assumed to fail simultaneously. (This provides slightly higher than 0.02% core inventory of noble gases and halogens released into the reactor coolant as specified in Regulatory Guide 4.2[4].

(5) An average of 1% of the noble gas activity and 1% of the halogen activity in the failed fuel rods is in the plenum, and these are the quantities released to the coolant[1].

(6) The equilibrium fission product activity in the core at the time of the accident is computed by use of the ORIGEN code[3]. The fission product inventory reflects the assumed 1000 days at

design power followed by a decay period of one minute. The one-minute assumption results in the decay of the very short-lived fission products which contribute significantly to the fission product inventory in the fuel, but are insignificant as far as plenum activity is concerned.

(7) The main steamline isolation valves are assumed to receive an automatic closure signal 0.5 second after the released activity reaches the main steamline radiation monitors and to be fully closed 5 seconds later. Thus, contaminated steam is assumed to flow into the system beyond the isolation valve for a total of 5.5 seconds.

(8) All of the noble gas activity is assumed to be released to the steam space. None is retained in the reactor coolant.

(9) One percent of the halogens in the reactor coolant is assumed to be released into the steamline. Ten percent of the halogens in the main condenser at a given time is assumed to be released to the atmosphere at the rate of 0.5% per day for 16 hours after the main condenser pressure rises to ambient (assumed to be 8 hours after the accident)[4].

(10) All the noble gas activity is assumed released to the main condenser and is available for release to the environment at a constant leak rate of 0.5% per day.

(11) The noble gases and halogens are released from the main condenser to the main condenser room below the operating floor and are removed from the turbine building via the ventilation stack at a rate of 3 air changes per hour[1].

(12) Main condenser room free volume was assumed to be 20,000 ft^3.

Based on the above assumptions and a radial power peaking factor of 1.5[4], the radionuclide activities in the main condenser and radionuclide concentrations in the room housing the main condenser are as shown in Table IV.

TABLE IV

RADIOACTIVITY RELEASED FROM FLOW BLOCKAGE ACCIDENT

Radionuclide	Main Condenser Radioactivity (8 hr decay) (Ci)	Radioactivity in the Main Condenser Room (μCi/cc) — Time After Accident: 8 hr + 10 min	9 hr	10 hr
Kr-83m	4.6	2.1(-7)[a]	3.7(-7)	2.7(-7)
Kr-85m	1.4(1)	6.7(-7)	1.4(-6)	1.3(-6)
Kr-85	2.6	1.2(-7)	3.0(-7)	3.2(-7)
Kr-87	1.2	5.3(-8)	8.3(-8)	5.0(-8)
Kr-88	1.9(1)	9.0(-7)	1.8(-6)	1.5(-6)
Xe-131m	1.6	7.6(-8)	1.8(-7)	1.9(-7)
Xe-133m	9.9	4.8(-7)	1.1(-6)	1.2(-6)
Xe-133	4.2(2)	1.9(-5)	4.8(-5)	5.0(-5)
Xe-135m	5.0(1)	2.3(-6)	5.3(-6)	5.0(-6)
Xe-135	1.8(2)	8.7(-6)	1.9(-5)	1.9(-5)
Br-83	1.8	8.1(-8)	1.5(-7)	1.2(-7)
I-131	2.2(2)	1.0(-5)	2.5(-5)	2.7(-5)
I-132	2.8(1)	1.3(-6)	2.3(-5)	1.8(-6)
I-133	4.0(2)	1.9(-5)	4.6(-5)	4.6(-5)
I-135	1.6(2)	7.6(-6)	1.7(-5)	1.6(-5)
Total	1.5(3)	7.0(-5)	1.9(-4)	1.7(-4)

[a] Example: $2.1(-7) = 2.1 \times 10^{-7}$.

III. ACCIDENT CLASS 6.0 – REFUELING ACCIDENTS (In Containment)

Accident 6.1 – Fuel Bundle Drop (In Containment)

1. INTRODUCTION

During a refueling cycle, fuel bundles can be relocated within the core, or they can be removed to a canal awaiting shipment to a reprocessing or storage facility. Fuel handling equipment and operating procedures are designed to minimize the possiblity of damaging fuel. The possibility, although remote, exists for the release of fission products at any time irradiated fuel is being handled.

2. POSTULATED CAUSES

During refueling operations, the primary containment head and reactor vessel head are removed. With the primary containment open and the reactor vessel head removed, any radioactive material released as a result of fuel failure is available for transport directly to the containment atmosphere. Refueling interlocks prevent an inadvertent criticality during refueling operations. The most likely refueling accident is fuel damage by mechanical means. For analysis of this class of accident, one fuel bundle is assumed to fall onto the top of the reactor core from the maximum height allowed by the fuel handling equipment. The maximum case postulates failure of all rods in the dropped bundle plus a conservatively high number of rods in the struck bundles (total of 171 rods). The realistic case postulates failure of only one row of 8 rods (pins) in the dropped bundle (per accident 6.1 of Reference 4). Accident 6.2 (of Reference 4) is not analyzed because it is considered intermediate in severity between the two cases considered (63 rods).

3. ACCIDENT DESCRIPTION

3.1.1 Fuel Bundle Drop Accident. A fuel bundle is dropped onto the reactor core from the maximum height available to the refueling equipment, 30 feet. The impact velocity is 40 ft/sec. The falling fuel bundle acquires a kinetic energy of 30,000 ft-lb which is dissipated in three impacts. Tests with point loads show that each fuel rod can absorb about 1 ft-lb bending load prior to cladding failure[1]. The impact forces will cause cladding failure in the dropped bundle and the struck bundles. The fission gases in the plenum are assumed to be completely released from failed fuel rods. The noble gases will migrate immediately through the water blanket over the core to the building atmosphere, while the halogens will be largely absorbed. As soon as high radiation levels are detected in the reactor building exhaust plenum, the normal ventilation system will be isolated and the Standby Gas Treatment System (SGTS) will be actuated[1].

4. SEQUENCE OF CONTAINMENT ISOLATION FOLLOWING THE POSTULATED ACCIDENT

Mark III containment is assumed for this analysis[1]. The containment pool is assumed to be filled with water for refueling operations. Following the accident, radioactive gas is released from the surface of the pool directly to the operating and refueling area atmosphere. The radioactive gas diffuses into the air above the pool, and from there it is drawn into the air exhaust ducts of the upper containment ventilation system. The ventilation exhaust radiation monitoring system consists of a number of radiation monitors arranged to monitor the activity level of the air exhausted from the containment. Upon detection of high radioactivity concentration levels in the containment atmosphere, the normal ventilation systems for the containment are automatically routed to the Standby Gas Treatment System (SGTS). The SGTS minimizes the exfiltration of contaminated air from the containment building.

5. EFFLUENT RELEASE PATHWAY

5.1.1 Point of Maximum Concentration. The released radioactive gases will rise from the reactor vessel through the pool water into the operating and refueling area atmosphere directly above the reactor vessel. The refueling platform must be in position above the reactor vessel in order for the postulated accident to occur; therefore, it is assumed to be the point of maximum concentration. The contaminated air will be exhausted by the containment ventilation system.

5.1.2 Principal Point of Environmental Release. The SGTS will remove contaminated air from the operating and refueling area at a rate of 5000 cfm[1]. During a refueling accident, both of the redundant SGTS units can be operated for a total capacity of 10,000 cfm, provided this does not exceed allowable release rates in the plant technical specifications. The point of release will be at the top of the plant vent which may vary in exact location from plant to plant. The release point is typically about 175 feet above grade[1].

6. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

(1) The plenum activity of all damaged fuel rods is released into the water[4].

(2) Reactor fuel has an irradiation time of 1000 days at design basis power (3800 MWt)[1]. Core fission product source terms were derived by use of the ORIGEN code[3].

(3) An average of 1% of the noble gas activity and 1% of the halogen activity in the core is located in the fuel plenums where it is available for release. The damaged rods have been located in the core where the radial power peaking factor is 1.5[4].

(4) Ventilation exhaust rate from the operating and refueling area at the time of the accident is 18,000 cfm[1].

(5) No particulate fission products are assumed to be released from the fuel.

(6) The noble gases are assumed to instantly traverse the column of water with no absorption and are released to the operating and refueling area atmosphere.

(7) Instantaneous equilibrium is achieved between the air and water for halogen concentrations. An iodine decontamination factor of 500 is assumed for water[4]. Effects of plateout and fallout are neglected.

(8) The refueling cavity liquid volume is 3.0×10^4 ft^3, and the containment air volume is 1.0×10^6 ft^3[1]; 100% building mixing is assumed for calculation of peak containment concentrations. Volume of air above the pool is assumed to be 10% of containment air volume.

(9) Only peak concentrations of radioactivity in containment were calculated. Radioactivity removal by the SGTS was not considered in this evaluation. Releases through the SGTS can be calculated by applying plant-specific decontamination factors and release parameters.

(10) Assuming a 5-second closure time for the 42-inch isolation valves and allowing 0.5 second for signal processing, contaminated air would be released to the atmosphere at a rate of three air changes per hour for 5.5 seconds before isolation.

6.1 Case 1 – Maximum Release Case

This accident is based on a fuel bundle being dropped from a height of 30 feet. No credit is taken for the drag forces resulting from falling through water.

(1) A 24-hour decay time is assumed to account for the minimum time required to remove the reactor vessel head after reactor shutdown[1].

(2) Kinetic energy of dropped rod is 30,000 ft-lb (drop from maximum height allowed by equipment).

(3) None of the energy associated with the dropped bundle is absorbed by fuel (UO_2)[1], i.e., all of the kinetic energy is absorbed by the fuel cladding and other core structural materials.

(4) Four fuel bundles are struck by the dropped fuel on its first impact which dissipates 80% of the kinetic energy of the dropped bundle.

(5) The second impact consists of the broad side of the dropped bundle impacting approximately 22 additional bundles, dissipating 19% of the initial energy.

(6) No fuel rods fail on the third or subsequent impacts (about 1% of the initial energy).

(7) The dropped bundle absorbs 50% of the impact energy causing failure of all 63 fuel rods in that bundle.

(8) All the energy absorbed by the struck bundles is concentrated in the rods that fail (worst possible case).

(9) Fuel rod failure in compression requires 250 ft-lb of energy, but only 1 lb-ft is required for bending mode failures.

(10) The eight tie rods in each struck bundle are susceptible to bending mode failure, but the remaining rods are held rigidly in place so they experience only compressive loads. Consequently, 32 rods in the struck bundles fail due to bending loads upon the initial impact.

(11) Approximately 12,000 ft-lb of energy from the initial impact is available to cause 48 failures[a] in compression among the remaining fuel rods (12,000/250).

(12) The second impact dissipates 5700 ft-lb of energy, half of which is spent in the core bundles. Two of the struck bundles are subject to bending failure of 16 additional tie rods.

(13) The remaining energy is available to cause 12 compressive failures (0.5 x 5700/250) resulting in a total of 171 failed rods.

The radioactivity released, based on the above assumptions, is summarized in Table V.

TABLE V

RADIOACTIVITY RELEASED BY FUEL BUNDLE DROP ONTO CORE (Maximum Case)[a]

Radionuclide	Radioactivity Released from Pool (Ci)	Concentration Above Pool (μCi/cc)	Radioactivity Released to Environment (Ci)	Peak Concentration in Containment (μCi/cc)
Kr-83m	1.9	6.8(-4)[b]	3.2(-2)	6.7(-5)
Kr-85m	4.0(2)	1.4(-1)	6.5	1.4(-2)
Kr-85	7.2(1)	2.5(-2)	1.2	2.5(-3)
Kr-87	3.4(1)	1.2(-2)	5.6(-1)	1.2(-3)
Kr-88	5.4(2)	1.9(-1)	8.6	1.9(-2)
I-131	1.2(1)	4.3(-3)	2.0(-1)	4.2(-4)
I-132	1.6(1)	5.8(-3)	2.7(-1)	5.7(-4)
I-133	1.8(1)	6.5(-3)	3.0(-1)	6.4(-4)
I-134	1.8(-1)	6.4(-5)	3.0(-3)	6.3(-6)
I-135	9.0	3.2(-3)	1.5(-1)	3.1(-4)
Xe-131m	4.4(1)	1.6(-2)	7.3(-1)	1.5(-3)
Xe-133m	2.8(2)	9.7(-2)	4.5	9.6(-3)
Xe-133	1.1(4)	4.1	1.9(2)	4.0(-1)
Xe-135m	1.4(3)	4.9(-1)	2.3(1)	4.9(-2)
Xe-135	5.2(3)	1.8	8.5(1)	1.8(-1)
Total	1.9(4)	6.9	3.2(2)	6.8(-1)

[a] 171 failed fuel pins.

[b] Example: 6.8(-4) = 6.8 x 10^{-4}.

[a] This ignores energy absorption by core structure which can exceed 50%.

6.2 Case 2 – Realistic Case

This case conforms in general with Accident 6.1 of Reference 4.

(1) One week decay time elapses before the accident occurs[4].

(2) One row of fuel pins (8 pins) releases the activity of the fuel plenum into water[4].

(3) The iodine decontamination factor for water is 500[4].

The radioactivity released, based on the above assumptions, is tabulated in Table VI.

TABLE VI

RADIOACTIVITY RESULTING FROM FAILURE OF EIGHT (8) FUEL PINS DURING FUEL HANDLING (Realistic Case)

Radionuclide	Radioactivity Released from Pool (Ci)	Concentration Above Pool (μCi/cc)	Radioactivity Released to Environment (Ci)	Peak Concentration in Containment (μCi/cc)
Kr-85	3.4	1.2(-3)[a]	5.5(-2)	1.2(-4)
I-131	3.2(-1)	1.1(-4)	5.3(-3)	1.1(-5)
I-132	1.8(-1)	6.5(-5)	3.0(-3)	6.4(-6)
I-133	4.4(-3)	1.5(-6)	7.2(-5)	1.5(-7)
Xe-131m	2.0	6.9(-4)	3.2(-2)	6.8(-5)
Xe-133m	2.5	8.8(-4)	4.1(-2)	8.6(-5)
Xe-133	2.6(2)	9.2(-2)	4.3	9.1(-3)
Xe-135	4.7(-3)	1.6(-6)	7.7(-5)	1.6(-7)
Total	2.7(2)	9.5(-2)	4.4	9.4(-3)

[a] Example: $1.2(-3) = 1.2 \times 10^{-3}$.

IV. ACCIDENT CLASS 7.0 – SPENT FUEL HANDLING ACCIDENT

Accident 7.1 – Fuel Bundle Drop in Fuel Storage Pool (Fuel Building)

Accident 7.2 – Heavy Object Drop Onto Fuel Rack

1. INTRODUCTION

The spent fuel is removed from the containment fuel storage pool to the fuel building storage pool via the fuel transfer tube and fuel transfer pool. When the fuel is being moved, it is kept underwater suspended from the refueling platform or the fuel handling platform. Any time the fuel is being handled during this operation, the potential exists for an accident to occur, resulting in release of fission products to the environment. For this reason, the radiological consequences of damage to fuel bundles were evaluated.

2. POSTULATED CAUSES

After the fuel bundles are lowered through the transfer tube into the fuel transfer building pool, they are moved through a gate into the fuel storage pool where they are placed in storage racks to await final disposition. The fuel storage racks are arranged to preclude the possibility of a criticality accident regardless of fuel bundle location below the fuel handling platform. Travel limits on the hoist prevent raising the fuel closer than 8 feet below the surface of the water in the storage pool. Clearance between the bottom of a fuel bundle suspended from the fuel handling platform and the top of the fuel in the storage racks cannot exceed 2.5 feet. The facility is designed so that no more than one fuel bundle can be handled at a time[1].

Accident conditions may result from an earthquake, accidental equipment drop, or damage caused by horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting gear. The fuel racks are designed to seismic Category I. The normal depth of water is about 25 feet above the stored fuel. Administrative controls prevent moving of objects above spent fuel in the pool. The fuel building and fuel storage facilities are designed to seismic Category I requirements. Building superstructure is also designed for protection against airborne missiles under tornado conditions[1].

The fuel storage racks are typically arranged in rows of 10 fuel bundles spaced to prevent criticality and bolted to the wall of the fuel storage pool. Each fuel bundle has an 8 x 8 array of fuel pins including one water rod. Eight of the 63 fuel pins in a bundle are tie rods, which are loaded to hold the remaining 55 pins rigidly in place. The tie rods are susceptible to failure in the bending mode, but the remaining rods are protected from the bending mode of failure. Tests with concentrated point loads applied to fuel rods show that in the bending mode only about 1 ft-lb of energy is necessary to cause cladding failure[1]. It takes about 250 ft-lb before cladding failure occurs in the compression mode[1].

The mechanical and administrative restrictions on equipment operation over and around the fuel pool preclude the possibility of a criticality accident in the pool. The most likely causes of a spent fuel accident are as follows:

(1) Drop of a fuel bundle or other heavy object during handling, in or over the fuel pool

(2) Movement of the fuel handling equipment horizontally while still engaged to a fuel bundle in the storage rack.

Either of these accidents could result in the release of radioactive products in the fuel building as a result of a cladding breach of individual fuel pins.

3. ACCIDENT DESCRIPTION

The fuel storage pool is located in the fuel building adjacent to the reactor building. Fuel is transferred from the reactor building to the fuel building through a fuel transfer tube. Lowering the carrier is done by means of a winch and cable system. The carrier is fitted with an insert for holding two fuel bundles.

Three accidents are evaluated:

(1) A maximum release case of a fuel bundle drop onto the fuel storage racks

(2) A realistic case of fuel bundle failure in bending due to horizontal movement of fuel handling equipment while coupled to a fuel bundle that is in a storage rack

(3) An intermediate release case for a heavy object drop onto the fuel storage racks.

In each case, radiation monitors in the fuel building ventilation system will detect the radiation and automatically initiate isolation of the building ventilation system and activation of the Standby Gas Treatment System (SGTS). Radioactivity released from the accident will be largely retained within the fuel building except for the short period of exhaust through the ventilation system before it is isolated.

4. EFFLU[illegible]VAY

4.1.1 Point of Maximum Co[illegible]leased radioactive gases will rise vertically through the water to the up[illegible] area where it will diffuse through the open area as it migrates toward the ven[illegible] sy[illegible]em exhaust intakes. The point of maximum concentration would be the und[illegible] fuel handling platform. The air conditioning system recirculates the air and [illegible] mixing of the radioactive gases with the building atmosphere.

4.1.2 Principal Point of Environmental Release. Whether the radioactive effluent is exhausted by the fuel building ventilation system or by the SGTS, [illegible] is finally released to the environment through a plant vent. For a BWR/6, the common point of release is a vent located at the top of the containment dome.

The release through the fuel building ventilation system contains all the radionuclides released to the building atmosphere, but the release through the SGTS will have removed 99% of the iodines[4]. Typically, the release point is about 175 feet above plant grade.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

(1) Fuel has an average irradiation time of 1000 days at the design basis power (3800 MWt)[1] and a radial power peaking factor of 1.5.

(2) An average of 1% of the noble gas activity and 1% of the halogen activity in the fuel rods has migrated to the plenum where it is available for release in the case of a cladding failure[4].

(3) No solid fission products are released[1].

(4) Iodine decontamination factor in water is 500[4].

(5) Radioactivity removal by the SGTS is ignored. Only peak containment concentrations were calculated.

(6) High radiation levels in the exhaust will isolate the fuel building ventilation system[1].

(7) Assuming a 5-second closure time for the 42-inch isolation valves, and allowing 0.5 second for signal processing, contaminated air would be released to the atmosphere at a rate of three air changes per hour for 5.5 seconds before isolation.

(8) Building volume is 700,000 ft^3, and 100% building mixing is assumed after building isolation. Calculation of release to the environment during 5.5 seconds assumes 30% building mixing and 200,000 ft^3 effective volume.

5.1 Case 1 – Fuel Bundle Drop in Fuel Storage Pool – Maximum Release Case

This case corresponds to the accident described in Reference 1:

(1) A channeled fuel bundle is dropped from the maximum achievable height of 2.5 feet onto fuel stored in the storage racks[1].

(2) The entire available potential energy at the time of the drop is available for dissipation in the fuel bundles involved in the accident. This assumption neglects any transfer of energy to the water from the dropped bundle[1].

(3) None of the energy is dissipated in fuel (UO_2); it is all dissipated in cladding or other structural material. An unchanneled fuel bundle consists of 76% fuel, 19% cladding, and 5% other material by weight; so this assumption is conservative[1].

(4) For a plastic impact, conservation of momentum results in a fractional energy dissipation during the first impact of 63% of the kinetic energy of the dropped bundle, based on the assumption that two stored bundles are struck initially.

(5) The second impact would likely be less direct. It is assumed that 20 bundles are struck by the then nearly horizontally oriented dropped bundle dissipating an additional 35% of the initial energy. This leaves only 2% of the energy to be dissipated in subsequent impacts.

(6) The kinetic energy acquired by the dropped bundle is 3000 ft-lb for a 600-pound fuel bundle dropped 2.5 feet.

(7) It is assumed that the dropped bundle is at a significant angle with the vertical so that the combination of impact force and momentum of the falling bundle introduces a bending couple which will cause all 63 fuel rods in the falling bundle to fail in the bending mode of failure[1].

(8) It is assumed that 50% of the energy dissipated from each impact is dissipated in the struck bundles, and the remainder in the dropped bundle. Consequently, 950 ft-lb (0.63 x 0.5 x 3000) of

energy is available to cause damage to the struck bundles. If the dropped bundle is tilted from the vertical at the time of impact, it can induce enough bending momentum in the struck bundles to cause failure of the 16 tie rods (worst case). The remaining fuel rods in the struck bundles are sufficiently well supported to be affected only by compression loads[1].

(9) Since it takes 250 ft-lb of energy to cause a cladding failure in compression and because the struck fuel bundle consists of 19% cladding and 5% other structural materials, four fuel rod failures are caused by compression on the first impact (950/250).

(10) Two additional rods can fail by compression on the second impact, and it is assumed that bending loads are imposed on an additional two bundles resulting in the failure of their 16 tie rods. The total number of failed rods is then conservatively estimated to be 63 + 16 + 4 + 16 + 2 = 101.

Fission product radioactivities (by isotope) released, based on one week of decay time and on the assumptions stated above, are presented in Table VII. The ORIGEN code was used to determine core fission product inventories[3].

TABLE VII

RADIOACTIVITY RELEASED BY FUEL BUNDLE DROP ONTO FUEL STORAGE RACK (Maximum Case)

Radionuclide	Radioactivity Released from Pool (Ci)	Concentration Above Pool (μCi/cc)	Radioactivity Released to Environment (Ci)	Peak Concentration in Containment (μCi/cc)
Kr-85	4.2(1)[a]	7.6(-3)	1.3	2.1(-3)
I-131	4.1	7.4(-4)	1.2(-1)	2.0(-4)
I-132	2.3	4.2(-4)	7.0(-2)	1.1(-4)
I-133	5.5(-2)	1.0(-6)	1.7(-3)	2.7(-6)
Xe-131m	2.5(1)	4.4(-3)	7.4(-1)	1.2(-3)
Xe-133m	3.1(1)	5.6(-3)	9.4(-1)	1.5(-3)
Xe-133	3.3(3)	5.9(-1)	9.9(+1)	1.6(-1)
Xe-135	5.9(-2)	1.1(-5)	1.8(-3)	2.8(-6)
Total	3.4(3)	6.1(-1)	1.0(2)	1.7(-1)

[a] Example: $4.2(1) = 4.2 \times 10^1$.

5.2 Case 2 – Horizontal Movement of Fuel Handling Equipment – Realistic Case (Corresponds in Severity to Accident 7.1 of Reference 4)

(1) While still attached to a fuel bundle located in the storage rack, the fuel handling platform is moved horizontally enough to introduce a small bending moment into the fuel bundle resulting in failure in the bending mode of the eight tie rods (equivalent to one row of eight rods)[4].

(2) One week decay time elapses before the accident occurs[4].

(3) Case 1 core fission product source terms are applicable to this accident.

(4) The activities released as a result of this accident are listed in Table VIII.

TABLE VIII

RADIOACTIVITY RELEASED BY MECHANICAL FAILURE OF EIGHT (8) FUEL PINS IN THE SPENT FUEL STORAGE POOL (Realistic Case)

Radionuclide	Radioactivity Released from Pool (Ci)	Concentration Above Pool (μCi/cc)	Radioactivity Released to Environment (Ci)	Peak Concentration in Contairment (μCi/cc)
Kr-85	3.4	6.1(-4)[a]	1.0	1.6(-4)
I-131	3.2(-1)	5.8(-5)	9.7(-2)	1.6(-5)
I-132	1.8(-1)	3.3(-5)	5.5(-2)	9.0(-6)
I-133	4.4(-3)	7.9(-7)	1.3(-3)	2.1(-7)
Xe-131m	2.0	3.5(-4)	5.9(-1)	9.6(-5)
Xe-133m	2.5	4.5(-4)	7.4(-1)	1.2(-4)
Xe-133	2.6(2)	4.7(-2)	7.8(1)	1.3(-2)
Xe-135	4.7(-3)	8.4(-7)	1.4(-3)	2.3(-7)
Total	2.7(2)	4.8(-2)	8.1(1)	1.3(-2)

[a] Example: $6.1(-4) = 6.1 \times 10^{-4}$.

5.3 Case 3 – Heavy Object Drop Onto Fuel Rack (Corresponds to Accident 7.2 of Reference 4)

(1) Ignoring administrative procedures, the 5-ton crane is assumed to be moving an object over the spent fuel in the fuel storage pool. The object breaks loose from the grapple and lands on the fuel storage rack causing failure of 63 fuel rods[4].

(2) Thirty days decay time has elapsed when the accident occurs[4].

(3) The source terms of Case 1 are used to determine release rates after extending the decay time to 30 days.

(4) Released activities are tabulated in Table IX.

TABLE IX

RADIOACTIVITY RELEASED BY HEAVY OBJECT DROP ONTO SPENT FUEL RACK
(Intermediate Case)

(63 Failed Rods and 30 Days Decay Time)

Radionuclide	Radioactivity Released from Pool (Ci)	Concentration Above Pool (μCi/cc)	Radioactivity Released to Environment (Ci)	Peak Concentration in Containment (μCi/cc)
Kr-85	2.6(1)[a]	4.8(-3)	7.9	1.3(-3)
I-131	3.5(-1)	6.3(-5)	1.1(-1)	1.7(-5)
I-132	1.1(-2)	1.9(-6)	3.2(-3)	5.2(-7)
Xe-131m	6.6	1.2(-3)	2.0	3.2(-4)
Xe-133	1.0(2)	1.8(-2)	3.0(1)	4.9(-3)
Total	1.3(2)	2.4(-2)	4.0(1)	6.5(-3)

[a] Example: 2.6(1) = 2.6×10^1.

V. ACCIDENT CLASS 8.0 – ACCIDENT INITIATION EVENTS CONSIDERED IN DESIGN BASIS EVALUATION IN THE SAFETY ANALYSIS REPORT

Accident 8.1 – Loss-of-Coolant Accidents

Accident 8.2(b) – Rod Drop Accident (BWR) Radioactive Material Released

Accident 8.3(b) – Steamline Breaks (BWR)

1. INTRODUCTION

The accidents covered in this class of events are extremely unlikely and generally require a large seismic event or simultaneous failure of two or more components. The piping systems in commercial reactors are built to Section III of the ASME Boiler and Pressure Vessel Code. This code requires strict controls and accounting for materials, fabrication, and installation. The pipe is also installed to withstand seismic events consistent with the history of the area. Control rod drop accidents require a combination of failure and bad judgment on the part of the operator for a maximum reactivity insertion to occur. The consequences of such accidents are the release of radioactive materials. For this reason these accidents are analyzed, although the probability of their occurrence is remote.

2. POSTULATED CAUSES

Three types of accidents are considered because of their potential severity. They are (a) reactivity insertion accidents (rod drop), (b) loss-of-coolant accidents (pipe break), and (c) steamline breaks. Reactivity insertion accidents usually occur as a result of a component failure coupled with operator error. The other two accidents, involving piping failures, are typically associated with seismic activity, missile impact, underdesigned systems, or poor fabrication. Adherence to the code provisions effectively eliminates design or construction deficiences.

Although no likely cause can be identified, assumptions are made for the postulated accidents. A coolant line is assumed to be breached to form a double ended offset shear (the failure that will provide the least flow resistance). The second postulated accident is based on a single control rod drive multiple failure and failure of the operator to notice the lack of response as the rod drive is withdrawn. The rod is assumed to be stuck in its fully inserted position; the drive is fully withdrawn; then, by some means, the rod is freed and drops to the fully withdrawn position, resulting in a large reactivity insertion. The third postulated accident assumes a steamline break.

3. ACCIDENT DESCRIPTION

Each of these accidents has a different effect on the plant, and they are grouped together only because of their improbability of occurrence and their severity. Because of dissimilarities, each will be discussed separately.

3.1 Loss-of-Coolant Accident

3.1.1 Large Pipe Break. A double ended recirculation line break occurs. The reactor is operating at 3800 MWt power with water level at the scram setpoint at the time of the accident. Normal power to the recirculation pumps is assumed to be lost at the time of the accident. Core flow is maintained at a relatively high level by the rotational energy of the recirculation pump in the unbroken line until the initial kinetic energy of the pump has been dissipated. The reactor pressure vessel depressurizes in approximately 50 seconds[1]. The emergency core cooling system (ECCS) begins delivering flow to the vessel within 30 seconds after the accident. The ECCS provides coolant soon enough and in sufficient quantities to prevent fuel melt. The peak cladding temperature has been calculated to be less than 2200°F[1]. Some cladding loss occurs at this temperature because it is near the threshold temperature for metal-water reactions for zirconium, but fuel rod failure does not occur.

The coolant that escapes through the pipe break is contained within the drywell where it flashes to steam causing the pressure in the drywell to increase. The increasing pressure differential between the drywell and the containment forces water from the drywell annulus through the vent holes and into the suppression pool where it is condensed. The radioactive gases that accompany the steam into the suppression pool either escape to the containment atmosphere or are scrubbed by the water.

3.1.2 Small Pipe Break. No mechanism has been identified for the initiation of this accident, but a severe break is assumed in a 6-inch recirculation suction pipe. The plant protection system scrams the reactor and the ECCS maintains the peak cladding temperature below 1500°F[1]. Consequently no breach of the cladding occurs, and fission product release is limited to the activity in the water.

3.2 Control Rod Drop Accident

Before the control rod drop accident is possible, the following sequence of events must occur:

(1) Complete rupture, breakage, or disconnection of a fully inserted control rod drive from its cruciform control blade at or near the coupling

(2) Sticking of the blade in the fully inserted position as the rod drive is withdrawn

(3) Drop of the control rod to the rod drive position.

This unlikely set of circumstances makes possible the rapid removal of a control rod. Drop of the rod results in a high local k_{eff} in a small region of the core. For large, loosely coupled cores, this results in a highly peaked power distribution and subsequent shutdown mechanisms. Significant shifts in the spatial power generation occur during the course of the excursion. Therefore, the method of analysis must be capable of accounting for any possible effects of the power distribution shifts. The sequence of events is as follows[1].

The reactor is operating at 50% control rod density pattern with rod pattern control system (RPCS) operational. The control rod which provides maximum incremental worth becomes decoupled. The operator selects and withdraws the control rod drive of the decoupled rod along with the other control rods assigned to the RPCS group or gang such that proper core geometry for maximum control rod worth exists. The decoupled control rod then sticks in the fully inserted position. The control rod becomes unstuck and drops at the nominal measured velocity (plus 3 standard deviations). The reactor then goes on a positive period and the initial power burst is terminated by Doppler reactivity feedback. A 120% power signal scrams the reactor, which terminates the accident sequence.

3.3 Steamline Break Accident

A steamline break can result in the release of radioactive materials outside the containment. No mechanism has been identified to cause this accident, but it is analyzed because of its potential consequences. The steamline is assumed to fail instantly providing a direct flow path to atmosphere. Steam flows from the opening for 5 seconds after an isolation signal is received. After isolation, the steam continues to flow until pressure is equalized between the steamline and the building atmosphere. The leak detection system automatically initiates isolation of the steamline when its preset limits are exceeded. The leak detection system monitors room temperature, flow rate, and liquid levels. Flow rates in floor drains are monitored as part of the leak detection system. Five seconds after the leak is detected, the line is assumed to be isolated.

4. EFFLUENT RELEASE PATHWAY

4.1 Loss-of-Coolant Accident (LOCA)

4.1.1 Point of Maximum Concentration. The activity is released to the drywell through the pipe break. Pressure rises in the drywell primarily due to the introduction from the pipe break of superheated water which flashes to steam. Any increase in pressure in the drywell is relieved by flow into the suppression pool through the system of horizontal vents that connect the drywell and containment pool volumes. The steam is largely condensed by the suppression pool water (initially less than 100°F) but all of the noble gases and some of the halogens are assumed to pass into the containment atmosphere. The noncondensible gases are retained in the free air volume inside the containment vessel until removed by the SGTS. The point of maximum concentration is in the containment directly over the suppression pool[1].

4.1.2 Principal Point of Environmental Release. The SGTS is used to clear the containment after a LOCA. The remaining gaseous fission products are released to the environment through the plant vent. This analysis calculates only peak containment concentrations. Removal of activity by the SGTS is ignored.

4.2 Rod Drop Accident

4.2.1 Point of Maximum Concentration. The rod drop accident will not cause a breach of the coolant system. Consequently, any radioactive gases released to the coolant either stay in the coolant (halogens) or carry over into the steam where they are transported to the main condenser. The steam system is assumed to be isolated within 5.5 seconds of the detection of high activity levels in the coolant system, with the result that the off-gas treatment system does not remove the radioactive gases from the main condenser. It is assumed that approximately 8 hours are required after steamline isolation for the main condenser pressure to reach ambient level; consequently, no appreciable release of activity from the main condenser occurs during this period. After ambient pressure is attained in the main condenser, the activity is released from the main condenser to the region below the operating floor[1].

4.2.1 Principal Point of Environmental Release. The turbine building and systems contained therein reflect the plant designer's preferences and experience. Consequently, turbine buildings and turbine building ventilation systems are plant-unique features which cannot be treated generically. However, it is assumed that the activity is removed from the turbine building via the ventilation exhaust system, which vents to the atmosphere at the turbine building roofline. The building is assumed to have a free volume of 1.0×10^6 ft^3.

4.3 Steamline Break

4.3.1 Point of Maximum Concentration. The steamline is assumed to break outside the containment to maximize the extent of the release. It is further assumed that the release occurs in the turbine building which is assumed not to be connected to the SGTS. The point of maximum concentration is in the pipe tunnel in the turbine building.

4.3.2 Principal Point of Environmental Release. The radioactive effluent will diffuse into the turbine building through the pipe tunnel penetrations. It will be pulled into the ventilation exhaust ducts for final environmental release at the turbine building roof level (50-foot elevation)[1].

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Loss-of-Coolant Accident

5.1.1 Maximum Release Case (Large Pipe Break). The following operating conditions were selected or assumed, to maximize severity of results:

(1) Normal power is assumed to be lost at the time of the break.

(2) The recirculation line fails instantaneously in the form of double ended offset shear, permitting discharge from both sides of the break.

(3) Containment isolation is complete at 5.5 seconds after the accident[1].

(4) No radioactivity removal by the SGTS is assumed. Only peak containment concentrations are calculated.

(5) The peak fuel cladding temperature is 2200°F which is insufficient to cause cladding perforation[1]. No fuel damage occurs, but 0.2% of the core inventory of halogens and noble gases are released to the drywell.

(6) Drywell volume is 276,000 ft^3.

(7) Containment volume is 10^6 ft^3.

(8) Initial drywell pressure is 15.5 psia.

(9) Total flow area of the break is 2.5 ft^2.

(10) Radioactivity inventory in the primary coolant is based on a release rate of 380,000 μCi/sec (30-minute decay) to the coolant.

(11) Fifty percent building mixing is assumed within the containment[2].

(12) A reduction factor of 0.2 is used in the drywell to account for the scrubbing of iodine by sprays and condensation.

(13) Reactor vessel initial pressure is 1060 psia.

(14) The entire coolant system activity is released to the drywell, and subsequently to the containment, except for the scrubbing of halogens by the process of condensation, drywell spray, and passage through the suppression pool.

The radioactivity released as a result of the accident is listed by nuclide in Table X. These values are based on core inventories calculated by use of the ORIGEN code[3].

TABLE X

RADIOACTIVITY RELEASED BY A BWR LOSS-OF-COOLANT ACCIDENT
(Maximum Case)

Radionuclide	Radioactivity Released to Containment (Ci)	Peak Containment Concentrations (μCi/cc)	Radioactivity Released to Environment Before Isolation (Ci)
Kr-83m	5.6(2)[a]	2.2(-1)	4.0(-2)
Kr-85m	1.8(3)	7.0(-1)	1.2(-1)
Kr-85	9.1(1)	3.6(-2)	6.4(-3)
Kr-87	3.4(3)	1.3	2.4(-1)
Kr-88	4.9(3)	2.0	3.5(-1)
Kr-89	5.1(3)	2.0	3.6(-1)
Kr-90	2.0(3)	7.9(-1)	1.4(-1)
Kr-91	8.0(1)	3.2(-2)	5.6(-3)
Xe-131m	5.6(1)	2.4(-2)	3.9(-3)
Xe-133m	3.5(2)	1.4(-1)	2.5(-2)
Xe-133	1.5(4)	5.8	1.0
Xe-135m	3.9(3)	1.6	2.8(-1)
Xe-135	4.9(3)	2.0	3.4(-1)
Xe-137	1.3(4)	5.1	9.1(-1)
Xe-138	1.3(4)	5.3	9.3(-1)
Xe-139	4.5(3)	1.8	3.2(-1)
Xe-140	5.9(2)	2.4(-1)	4.2(-2)
Br-83	1.1(2)	4.5(-2)	7.9(-3)
Br-84	2.8(2)	1.1(-1)	2.0(-2)
Br-85	2.9(2)	1.1(-1)	2.0(-2)
I-131	1.6(3)	6.2(-1)	1.1(-1)
I-132	2.2(3)	8.9(-1)	1.6(-1)
I-133	2.9(3)	1.2	2.1(-1)
I-134	3.3(3)	1.3	2.3(-1)
I-135	2.6(3)	1.0	1.8(-1)
Total	8.6(4)	3.4(1)	6.1

[a] Example: 5.6(2) = 5.6 x 10^2.

5.1.2 Realistic Case (Small Pipe Break)

(1) A 6-inch core spray line pipe breaks circumferentially inside the drywell. The broken ends are displaced to permit unrestricted flow from each end of the pipe[4].

(2) Containment isolation is complete at 5.5 seconds after the accident[1].

(3) Containment volume is 1.0×10^6 ft^3[1].

(4) The peak fuel rod cladding temperature is 1500°F. No cladding failure occurs[1].

(5) Fifty percent building mixing is assumed[4].

(6) Vessel pressure is 1060 psia and is maintained throughout the accident[1].

(7) No noble gases are transported to the drywell.

(8) Enthalpy of the leaking coolant is assumed constant at 530 Btu/lb[1]. The corresponding liquid temperature is 535°F and the density is 47 lb/ft^3. When this flashes to steam, the resulting mixture is 35% steam and 65% water.

(9) Based on the above assumptions, a liquid flow rate into the drywell of 250 lb/sec is computed.

(10) The drywell pressure rises at a rate of 0.6 psi/sec. When the drywell pressure reaches 2.75 psig above the containment pressure, i.e., 5 seconds after the pipe break, the top vent in the suppression tank clears. At this time the contents of the drywell begin to vent to the suppression pool where the steam condenses and the noncondensibles pass through to the containment.

(11) High radiation alarms and drywell pressure alarms alert the operator to the accident. He effects an orderly shutdown of the reactor. Reactor depressurization takes approximately 10 minutes[1].

(12) Based on the above assumptions, 77,000 pounds of coolant are transferred to the drywell. Thirty-five percent will flash to steam (27,000 pounds). The drywell retains 11,000 pounds of steam, and the remainder passes into the suppression pool. All the

drywell air is assumed to be expelled into the containment. Released activities are summarized (by radionuclide) in Table XI.

(13) A reduction factor of 0.2 is used in the drywell to account for the scrubbing of iodine by sprays and condensation.

TABLE XI

RADIOACTIVITY RELEASED FROM A BWR SPRAY LINE BREAK
(Realistic Case)

Radionuclide	Coolant Radioactivity (μCi/g)	Radioactivity Released to Containment (Ci)	Peak Concentration in Containment (μCi/cc)	Radioactivity Released to Environment Before Isolation (Ci)
Br-83	3(-3)[a,b]	2(-2)	8(-6)	1(-6)
Br-84	5(-3)	3(-2)	1(-5)	2(-6)
Br-85	3(-3)	2(-2)	8(-6)	1(-6)
I-131	5(-3)	3(-2)	1(-5)	2(-6)
I-132	3(-2)	2(-1)	8(-5)	1(-5)
I-133	2(-3)	1(-1)	6(-5)	1(-5)
I-134	5(-2)	3(-1)	1(-4)	2(-5)
I-135	2(-2)	1(-1)	6(-5)	1(-5)
Total		8(-1)	3(-4)	6(-5)

[a] Example: 3(-3) = 3 x 10^{-3}.

[b] Data accuracy to one significant digit.

5.2 Rod Drop Accident

5.2.1 Maximum Release Case. A control rod is assumed to drop completely out of the core. The resulting reactivity insertion causes failure of some fuel rods releasing radioactive gases to the coolant. The assumptions relative to the release of radioactive gases from the core as a result of this accident are as follows:

(1) The reactor has operated at design power for sufficient time to provide equilibrium core inventory conditions (1000 hours at full power).

(2) The release to the coolant consists of 0.025% of the core inventory of noble gas and 0.025% of the core inventory of halogens[4].

(3) A total of 1% of the halogens in the reactor coolant are released into the main condenser[4].

(4) The mechanical vacuum pump is assumed to be nonoperational during the course of the accident[4].

(5) The radioactivity in the steamline causes the main steam valve to be fully closed 5.5 seconds after the accident occurs. This number includes 0.5-second signal processing time and 5-seconds valve closure time. Full steam flow rate with contaminated steam (17×10^6 lb/hr) occurs for the full 5.5 seconds. A total of 26,000 pounds of contaminated steam reaches the main condenser representing 5% of the total inventory of coolant in the plant or the full normal inventory of steam.

(6) The reactor has been releasing 380,000 μCi/sec (measured at 30-minute decay) into the coolant for a period of 30 days prior to the accident.

(7) All the noble gas activity released into the coolant is released to the steam space and carried with the steam to the main condenser.

(8) Main condenser pressure rises to atmospheric pressure in 8 hours. After the 8-hour period, 10% of the halogens and all of the noble gases in the main condenser are available for leakage from the main condenser to the environment at 0.5% per day for the course of the accident (24 hours)[4].

(9) The point of maximum concentration is in the main condenser room below the operating floor.

Based on the above listed assumptions the concentrations of radioactivity are listed by nuclide in Table XII.

5.2.2 Realistic Case. This accident is the same as the maximum release case, except that the reactor power is at 10% when the rod drop accident occurs. All the other assumptions relating to the condition at the time of the accident are the same as for the maximum release case unless indicated:

(1) The peak enthalpy in the fuel is about 200 cal/gm and results in the failure of 60 fuel rods[1].

TABLE XII

RADIOACTIVITY CONCENTRATIONS IN MAIN CONDENSER ROOM FOLLOWING
A BWR ROD DROP ACCIDENT
(Maximum Case)

Radionuclide	Activity in Main Condenser (8 hr decay) (Ci)	Concentration in Main Condenser Room (μCi/cc)		
		8 hr + 10 min	9 hr	10 hr
Kr-83m	5.8(2)	2.7(-5)[a]	4.6(-5)	3.4(-5)
Kr-85m	1.8(3)	8.5(-5)	1.8(-4)	1.6(-4)
Kr-85	3.2(2)	1.6(-5)	3.7(-5)	3.9(-5)
Kr-87	1.5(2)	6.7(-6)	1.0(-5)	6.4(-6)
Kr-88	2.4(3)	1.1(-4)	2.3(-4)	1.8(-4)
Xe-131m	2.0(2)	9.6(-6)	2.3(-5)	2.5(-5)
Xe-133m	1.2(3)	6.0(-5)	1.4(-4)	1.5(-4)
Xe-133	5.2(4)	2.5(-3)	6.0(-3)	6.4(-3)
Xe-135m	6.3(3)	3.0(-4)	6.6(-4)	6.2(-4)
Xe-135	2.3(4)	1.1(-3)	2.5(-3)	2.5(-3)
Br-83	2.2	1.0(-8)	1.9(-8)	1.5(-8)
I-131	2.7(2)	1.3(-6)	3.2(-6)	6.1(-6)
I-132	3.5(1)	1.6(-7)	3.0(-7)	2.3(-7)
I-133	5.0(2)	2.5(-6)	5.7(-6)	5.8(-6)
I-135	2.0(2)	9.6(-7)	2.1(-6)	1.9(-6)
Total		4.3(-3)	9.8(-3)	1.0(-2)

[a] Example: $2.7(-5) = 2.7 \times 10^{-5}$.

(2) The reactor is at 10% power at the time of the accident, had just been reduced to that power, and is being returned to full power (3800 MWt).

(3) Based on the assumptions used, 0.0013% of the core inventory of noble gases and halogens are released from the failed fuel rods to the reactor coolant water.

(4) The steam flow to the main condenser is 10% of rated, and steam flows at this rate for 5.5 seconds (until the valve is fully closed).

(5) One percent of the halogens released to the water are assumed to enter the steamlines. Based on these assumptions, the amount of contaminated steam in the main condenser at isolation is 2600 pounds or 12% of the mass of steam in the reactor. This results in 1.6×10^{-4}% of the core inventory of noble gases and 1.6×10^{-6}% of the halogens reaching the main condenser.

(6) Ten percent of the halogens in the main condenser are assumed to be available for leakage from the main condenser at a volumetric rate of 0.5% per day.

(7) Leakage from the main condenser does not start until 8 hours after the accident and continues for 16 hours.

(8) Radioactivity entering the steam prior to the accident is at the rate of 60,000 μCi/sec (30-minute decay).

Based on these assumptions, radioactivity concentrations in the main condenser room are listed in Table XIII.

5.3 Steamline Break Accident

5.3.1 Main Steamline Break – Maximum Release Case. The worst case accident is a complete circumferential severance of a main steamline outside the secondary containment[1]. The assumptions relating to the release of radioactive gases from the coolant are as follows:

(1) The reactor is operating at 3800 MWt, and equilibrium core fission product conditions exist in the core.

(2) The release rate to the steam was 380,000 μCi/sec (measured at 30-minute decay) for 30 days prior to the accident.

(3) Isolation valves are fully closed 5.5 seconds after the accident.

(4) Fifty percent of the halogens in the fluid exiting the break are released to the steam tunnel atmosphere[4].

(5) The break is assumed to occur inside the steam tunnel, which has free volume of 16,300 ft^3.

(6) Based on the above assumptions, a total of 35,000 pounds of steam is released to the steam tunnel.

TABLE XIII

RADIONUCLIDE CONCENTRATIONS IN THE MAIN CONDENSER ROOM FOLLOWING A BWR ROD DROP ACCIDENT
(Realistic Case)

Radionuclide	Activity in Main Condenser (8 hr decay) (Ci)	Concentration in Main Condenser Room (μCi/cc)		
		8 hr + 10 min	9 hr	10 hr
Kr-83m	3.6	1.6(-7)[a]	2.8(-7)	2.1(-7)
Kr-85m	1.1(1)	5.1(-6)	1.1(-5)	9.9(-6)
Kr-85	2.0	9.7(-8)	2.3(-7)	2.5(-7)
Kr-87	9.5(-1)	4.2(-8)	6.4(-8)	3.9(-8)
Kr-88	1.5(1)	6.9(-7)	1.4(-6)	1.1(-6)
Xe-131m	1.2	6.0(-8)	1.5(-7)	1.5(-7)
Xe-133m	6.2	3.0(-7)	7.1(-7)	7.4(-7)
Xe-133	3.2(2)	1.6(-5)	3.7(-5)	3.9(-5)
Xe-135m	3.9(1)	1.8(-6)	4.1(-6)	3.9(-6)
Xe-135	1.4(2)	6.9(-6)	1.6(-5)	1.5(-5)
Br-83	1.4(-2)	6.4(-11)	1.2(-10)	9.4(-11)
I-131	1.7	8.3(-9)	1.9(-8)	2.1(-8)
I-132	2.2(-1)	9.9(-10)	1.9(-8)	1.5(-9)
I-133	2.6	1.2(-7)	2.8(-7)	3.0(-7)
I-135	1.3	5.8(-8)	1.3(-7)	1.2(-7)
Total		3.1(-5)	7.1(-5)	7.0(-5)

[a] Example: 1.6(-7) = 1.6×10^{-7}.

(7) The reactor pressure vessel depressurization results in a higher than normal halogen release rate to the steamline. The halogen activity in the steam is assumed to be 2% of the activity in the water on a weight basis[2].

Based on the above assumptions, the activity released to the steam tunnel is summarized in Table XIV.

5.3.2 Small Steam Pipe Break – Realistic Case. A 0.25 ft^2 break occurs in a steamline[4]. Except for the size of the break and the exceptions noted below, this accident is based on the same assumptions as the large pipe break. The assumptions specifically related to the release of radioactive gases from the coolant as a result of this accident are as follows:

TABLE XIV

RADIOACTIVITY RELEASED TO STEAM TUNNEL BY A MAIN STEAMLINE BREAK
(Maximum Case)

Radionuclide	Radioactivity Released (Ci)	Peak Concentration in Tunnel (μCi/ml)
Kr-83m	1.1(-1)[a]	2.4(-4)
Kr-85m	1.9(-3)	4.2(-4)
Kr-85	6.1(-4)	1.3(-6)
Kr-87	6.7(-1)	1.4(-3)
Kr-88	6.7(-1)	1.4(-3)
Kr-89	4.1	9.0(-3)
Kr-90	9.1	2.0(-2)
Kr-91	1.1(1)	2.4(-2)
Kr-92	1.1(1)	2.4(-2)
Kr-93	2.9	6.4(-3)
Kr-94	7.3(-1)	1.6(-3)
Kr-95	6.7(-2)	1.4(-4)
Xe-131m	4.7(-4)	1.0(-6)
Xe-133m	9.1(-3)	2.0(-5)
Xe-133	2.6(-1)	5.7(-4)
Xe-135m	8.5(-1)	1.8(-3)
Xe-135	7.3(-1)	1.6(-3)
Xe-137	4.7	1.0(-2)
Xe-138	2.8	6.1(-3)
Xe-139	9.1	2.0(-2)
Xe-140	9.7	2.1(-2)
Xe-141	7.9	1.7(-2)
Xe-142	2.3	5.0(-3)
Xe-143	3.8(-1)	8.3(-4)
Xe-144	1.8(-2)	3.9(-5)
Br-83	3(-3)[b]	5(-6)
Br-84	5(-3)	1(-5)
Br-85	3(-3)	5(-6)
I-131	5(-3)	1(-5)
I-132	3(-2)	5(-4)
I-133	2(-2)	9(-5)
I-134	5(-2)	1(-4)
I-135	2(-2)	5(-5)
Total	7.9(1)	1.7(-1)

[a] Example: $1.1(-1) = 1.1 \times 10^{-1}$.

[b] Data accuracy to one significant digit.

(1) The release rate to the steam prior to the accident was 60,000 μCi/sec (measured at 30-minute decay).

(2) The break is located so that no obstruction to flow through the break will exist.

(3) No appreciable depressurization of the reactor, and consequently no increase in release rate of radioactivity to the steam, occurs during the course of the accident.

(4) The steam flow rate through the break is estimated at 500 lb/sec for the duration of the accident (5.5 seconds). This rate is less than half the normal flow rate through a single main steamline (1180 lb/sec).

(5) The total mass of steam discharged through the break prior to isolation valve closure is 2700 pounds. The steam is discharged directly to the steam tunnel.

The activity released for the small steam pipe break is listed in Table XV.

TABLE XV

RADIOACTIVITY RELEASED BY A SMALL STEAM PIPE BREAK
(Realistic Case)

Radionuclide	Radioactivity Released (Ci)	Peak Concentration in Tunnel (μCi/cc)
Kr-83m	1.4(-3)[a]	3.0(-6)
Kr-85m	2.4(-3)	5.2(-6)
Kr-85	7.5(-6)	1.6(-8)
Kr-87	8.3(-3)	1.8(-5)
Kr-88	8.3(-3)	1.8(-5)
Kr-89	5.1(-2)	1.1(-4)
Kr-90	1.1(-1)	2.4(-4)
Kr-91	1.4(-1)	3.0(-4)
Kr-92	1.4(-1)	3.0(-4)
Kr-93	3.6(-2)	7.9(-5)
Kr-94	9.0(-3)	2.0(-5)
Kr-95	8.3(-4)	1.8(-6)
Xe-131m	5.9(-6)	1.3(-8)
Xe-133m	1.1(-4)	2.4(-7)
Xe-133	3.3(-3)	7.1(-6)
Xe-135m	1.1(-2)	2.3(-5)
Xe-135	9.0(-3)	2.0(-5)
Xe-137	5.9(-2)	1.3(-4)
Xe-138	9.5(-2)	7.6(-5)
Xe-139	1.1(-1)	2.4(-4)
Xe-140	1.2(-1)	2.6(-4)
Xe-141	9.8(-2)	2.1(-4)
Xe-142	2.9(-2)	6.2(-5)
Xe-143	4.8(-3)	1.0(-5)
Xe-144	2.3(-4)	4.9(-7)
Br-83	4(-5)[b]	8(-8)
Br-84	6(-5)	1(-7)
Br-85	4(-5)	8(-8)
I-131	6(-5)	1(-7)
I-132	4(-4)	8(-7)
I-133	3(-4)	5(-7)
I-134	6(-4)	1(-6)
I-135	3(-4)	5(-7)
Total	1.0	2.3(-3)

[a] Example: 1.4(-3) = 1.4×10^{-3}.

[b] Data accuracy to one significant digit.

PART II

PRESSURIZED WATER REACTORS

PART II

PRESSURIZED WATER REACTORS

VI. ACCIDENT CLASS 3.0 – RADWASTE SYSTEM FAILURE

Accident 3.2 – Release of Waste Gas Storage Tank Contents (PWR)

1. INTRODUCTION

In most pressurized water reactors (PWR), pressurized waste gas storage tanks are used to permit decay of accumulated radioactive fission product gases stripped from the primary coolant as a means of reducing or preventing the direct release of radioactive gases to the atmosphere.

Multiple waste gas storage tanks are provided at most plants to allow operating flexibility and to permit one or more tanks to be isolated from the rest of the system for extended periods of time. Most of the volume of the gas stored in the storage tanks is nitrogen cover gas displaced from the primary coolant purification system liquid holdup tanks. The radioactive components, which represent a small portion of the total volume of gas, are principally the noble gases (krypton and xenon isotopes) and trace quantities of halogens[5]. The waste gas storage tanks are usually located in the auxiliary building.

The possibility exists for the release of radioactive gases from the pressurized waste gas storage tanks either by leakage or by the gross failure of a tank (or attaching lines).

2. POSTULATED CAUSES

Although systems are designed to prevent accidental leakage or rupture of the pressure boundary, the failure of valve packing or small lines connected to the tanks can be postulated to cause release of radioactive waste gas. The release of gas by a gross tank rupture could be caused by mechanical damage, failure of the overpressurization system, or a hydrogen explosion within a tank (unless the tank is designed to withstand the effects of such an explosion).

3. ACCIDENT DESCRIPTION

Two postulated accidental releases of the contents of one gas storage tank are considered. The first mechanism is gross failure of the tank. The second mode of failure could occur by any one of several mechanisms, including an inlet or discharge pipe break, or valve malfunction.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The room or space in which the waste gas storage tanks are located is assumed to be the point at which the maximum concentration of airborne radionuclides would occur in case of leakage or failure. Multiple tanks may be installed in separate compartments for maintenance; however, such compartments are open to gas flow. For this study, a common hallway to the tanks was assumed as the location for calculation of the maximum postulated concentration.

4.1.2 Principal Point of Environmental Release. For the case of tank rupture, damage to or rupture of ventilation exhaust ducts is possible. In such an event, significant quantities of radioactive gases could reach many areas of the auxiliary building, but probably only on the level on which the waste gas storage tanks are located. For the more realistic release, the pressure rise associated with a release in the storage tank room would be insignificant, and rupture of or damage to the exhaust ducting would not be expected. For this study, it was assumed that the portion of gaseous effluent going through the auxiliary building ventilation system will pass through a HEPA filter before discharge near the top of the containment building at approximately 100,000 cfm (i.e., will be diluted by a factor of approximately 20 to 30 from other exhaust air inputs from the auxiliary building ventilation system).

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case

In the evaluation of the postulated accident assuming gross rupture of the waste gas storage tank, the fission product accumulation and release assumptions are as follows:

(1) The reactor has been operating at full power with a defective fuel rate such that a maximum of 150,000 curies of noble gas activity has accumulated in the waste gas storage tank prior to failure. The quantity is based on a generic accident analysis and is considered to be independent of either reactor design power level or an established fuel failure rate.

(2) The failure is assumed to occur immediately upon completion of the waste gas transfer (during which no decay is assumed) releasing the entire contents of the tank (4800 scf) to the area of the auxiliary building housing the waste gas storage tanks[2]. The assumption of the release of the noble gas inventory from a

single tank is based on a design which allows each waste gas storage tank to be isolated from all others[6]. Although rupture of a tank could result in rupture of ventillation ducts connected to the storage tank room, and flow from the rooms could occur, none is assumed in calculating the postulated concentrations. The room free volume, including that of the failed tank is assumed to be 20,000 ft^3.

Table XVI presents the calculated activities released, by radionuclide, and the maximum concentrations within the affected room resulting from the maximum postulated release of the contents of one waste gas storage tank.

TABLE XVI

RADIOACTIVITY RELEASED FROM RUPTURED WASTE GAS STORAGE TANK
(Maximum Case)

Radionuclide	Radioactivity Released (Ci)	Peak Room Concentration (μCi/cc)
Br-83	3.7	6.6(-3)[a]
Br-84	2.0	3.6(-3)
Br-85	2.3(-1)	4.1(-4)
I-130	1.6	2.9(-3)
I-131	2.1(2)	3.7(-1)
I-132	7.7(1)	1.4(-1)
I-133	2.9(2)	5.2(-1)
I-134	3.6(1)	6.4(-2)
I-135	1.5(2)	2.6(-1)
Kr-83m	1.6(2)	2.9(-1)
Kr-85m	8.6(2)	1.5
Kr-85	1.2(3)	2.1
Kr-87	4.7(2)	8.2(-1)
Kr-88	1.6(3)	2.7
Kr-89	3.9(1)	6.9(-2)
Xe-131m	8.6(2)	1.5
Xe-133m	1.7(3)	3.0
Xe-133	1.4(5)	2.5(2)
Xe-135m	1.0(2)	1.8(-1)
Xe-135	2.7(3)	4.8
Xe-137	7.0(1)	1.2(-1)
Xe-138	3.4(2)	6.0(-1)
Total	1.5(5)	2.7(2)

[a] Example: $6.6(-3) = 6.6 \times 10^{-3}$.

5.2 Case 2 – Realistic Case

In the evaluation of the more realistic release of fission products from a waste gas storage tank, a number of release assumptions differing from the postulated maximum case are made. The assumptions are related to the release of radioactive gases resulting from postulated failure of a one-inch line (which also would approximate any number of similar releases resulting from relief device, operator error, or valve malfunction):

(1) The reactor has been operating at full power with approximately 0.12% defective fuel[5]. The reactor has been shut down to a cold condition prior to the accident.

(2) All noble gases removed from the primary coolant are transferred to a single waste gas storage tank which subsequently is assumed to fail by failure of a one-inch line. Failure occurs 24 hours after reactor shutdown – a reasonable time for gas stripping of one primary coolant volume and completion of transfer of the gases to the storage tank.

(3) Although the release time is actually somewhat less than one minute, instantaneous release is assumed for purposes of concentration calculation.

(4) The release is uniformly mixed in the waste gas storage tank rooms and valve corridor in a volume of 20,000 ft^3.

Table XVII presents the calculated noble gas and halogen activities released and radioactivity concentrations expected within the affected tank room resulting from a postulated realistic accidental release from a waste gas storage tank.

TABLE XVII

RADIOACTIVITY RELEASED FROM RUPTURED WASTE GAS STORAGE TANK
(Realistic Case)

Radionuclide	Radioactivity Released (Ci)	Peak Room Concentration (μCi/cc)
I-130	1.3(-2)[a]	2.4(-5)
I-131	6.3	1.1(-2)
I-132	2.0	3.5(-3)
I-133	4.5	7.9(-3)
I-135	3.9(-1)	7.0(-4)
Kr-83m	2.3(-2)	4.0(-5)
Kr-85m	6.3(-1)	1.1(-3)
Kr-85	3.7(1)	6.6(-2)
Kr-88	1.3(-1)	2.3(-4)
Xe-131m	2.7(1)	4.8(-2)
Xe-133m	5.0(1)	8.9(-2)
Xe-133	4.3(3)	7.7
Xe-135m	2.8(-1)	5.0(-4)
Xe-135	3.3(1)	5.8(-2)
Total	4.5(3)	8.0

[a] Example: $1.3(-2) = 1.3 \times 10^{-2}$.

VII-A. ACCIDENT CLASS 5.0 – FISSION PRODUCTS TO PRIMARY AND SECONDARY SYSTEMS (PWR)

Accident 5.2 – Off-Design Transients That Induce Fuel Failure Above Those Expected and Steam Generator Leak

1. INTRODUCTION

A primary coolant pump locked rotor is probably the most commonly considered limiting fault or off-design transient, not covered in other accident classes, that potentially can induce fuel failure and cause release of fission products. The seizure of a primary coolant pump rotor would cause rapid reduction in available flow for core cooling, a consequent departure from nucleate boiling (DNB) in some of the fuel, a pressure rise in the primary coolant system, and if accompanied by leakage from the primary to secondary coolant system, would result in fission product release to the environment through the secondary coolant system steam relief valves or through the condenser vacuum system.

2. POSTULATED CAUSES

Failures of a primary coolant pump resulting in an instantaneously locked rotor have not been experienced in a reactor plant. Failures of prototype pumps which could have subsequently led to a locked rotor have occured; however, and it may be postulated that failure of bearings, material defects, loss of cooling water, and other events could initiate a seizure of a primary coolant pump rotor.

3. ACCIDENT DESCRIPTION

In the event of instantaneous seizure of a primary coolant pump rotor, flow through the affected primary coolant loop would be rapidly reduced, leading to initiation of a reactor trip on a low-flow signal.

Following initiation of the reactor trip, the low-flow condition would persist, and heat stored in the fuel rods would continue to be transferred to the primary coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators would be reduced, first because the reduced flow results in a decreased tube side film coefficient and subsequently because the primary coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the primary coolant in the reactor core, combined with reduced heat transfer in the steam generators, would cause a surge into the pressurizer, and a pressure increase would result throughout the primary coolant system. The surge into the pressurizer would compress the gas volume of the pressurizer, actuate the automatic spray system, open the power-operated relief valves, and open the pressurizer safety valves, in that sequence. The power-operated relief valves are designed for reliable operation and would be expected to function during the accident[6]. If steam flow to the turbine were stopped by the reactor scram, the secondary coolant system pressure would be expected to rise, similar to an accident considering loss of station auxiliary power, and steam would be released through the secondary coolant system steam relief valves or the condenser vacuum system or a combination of these.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The maximum airborne fission product concentration following a primary coolant pump seizure would occur at either the steam relief valve discharge or the condenser vacuum system discharge vent. These discharge locations are usually above the elevation of the turbine building roof.

4.1.2 Principal Point of Environmental Release. For the primary coolant pump seizure accident, the principal points of environmental release would be the steam relief valve discharge or the condenser vacuum system discharge vent. This evaluation assumes that all gaseous release is through the steam relief valve discharge.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Postulated Case

The assumptions and accident conditions for this case are taken from NUREG-0099 (Regulatory Guide 4.2, Rev. 2), Appendix I[4] and NUREG-0017[5].

(1) The primary coolant activity is based on operation with 0.5% failed fuel[4]. The reactor is assumed to have been operated at a core thermal power of 3800 MWt continuously for a period sufficient to establish equilibrium concentrations of the radionuclides in the primary coolant (620 days).

(2) All noble gases and 0.1% of the halogens in the steam reaching the condenser are assumed to be released by the condenser vacuum system[4].

(3) Equilibrium concentration of activity in the secondary coolant is based on an assumed primary-to-secondary leak of 100 lb/day of

primary coolant activity[5]. Equilibrium concentrations of radionuclides in the secondary coolant were taken from Tables 2-2 and 2-3 of NUREG-0017[5], with values extrapolated to a failed fuel rate of 0.5%.

(4) Releases were calculated for plants with once-through steam generators, U-tube steam generators with all-volatile water treatment, and U-tube steam generators with phosphate water treatment. All releases were similar except for the releases of larger quantities of halogens from a plant with phosphate water treatment.

Table XVIII presents the calculated radionuclide release concentration (Ci/10^3 lb of secondary coolant released as steam or vapor) from the steam relief valve discharge to the atmosphere following seizure of a primary coolant pump rotor for PWRs using U-tube steam generators with phosphate water treatment and for PWRs using once-through steam generators.

TABLE XVIII

RADIOACTIVITY RELEASED AT SECONDARY COOLANT SYSTEM STEAM RELIEF VALVES FOLLOWING LOCKED PRIMARY COOLANT PUMP ROTOR ACCIDENT

	Release (Ci/1000 lb secondary coolant)[a]	
Radionuclide	Once-through Steam Generator Plants	U-tube Phosphate Treatment System
Br-83	4.3(-12)[b]	2.8(-10)
Br-84	2.3(-12)	3.8(-11)
Br-85	2.6(-13)	3.8(-13)
I-130	2.5(-13)	4.7(-10)
I-131	2.5(-10)	2.1(-7)
I-132	8.9(-11)	2.1(-8)
I-133	3.4(-10)	1.2(-7)
I-134	4.2(-11)	1.1(-9)
I-135	1.7(-10)	2.6(-8)
Kr-83m	1.1(-8)	1.1(-8)
Kr-85m	5.9(-8)	5.9(-8)
Kr-85	7.9(-8)	7.9(-8)
Kr-87	3.0(-8)	3.0(-8)
Kr-88	1.0(-7)	1.0(-7)
Kr-89	2.6(-9)	2.6(-9)
Xe-131m	5.9(-8)	5.9(-8)
Xe-133m	1.2(-7)	1.2(-7)
Xe-133	9.5(-6)	9.5(-6)
Xe-135m	6.8(-9)	6.8(-9)
Xe-135	1.8(-7)	1.8(-7)
Xe-137	4.7(-9)	4.7(-9)
Xe-138	2.3(-8)	2.3(-8)
Total	1.0(-5)	1.1(-5)

[a] Discharged as steam.

[b] Example: 4.3(-12) = 4.3×10^{-12}.

VII-B. ACCIDENT CLASS 5.0 – FISSION PRODUCTS TO PRIMARY AND SECONDARY SYSTEMS (PWR)

Accident 5.3 – Steam Generator Tube Rupture

1. INTRODUCTION

The failure of a steam generator tube leads to an increase in contamination of the secondary coolant system due to leakage of primary coolant into the secondary system. Since the primary coolant system is normally contaminated with fission products due to operation with a limited amount of defective fuel and also because the fuel cladding contains "tramp" uranium, the discharge of activity due to a steam generator tube rupture can be significant. If the reactor continues operation on failure of a tube, the discharge of radioactive gases is through the condenser vacuum system. If, however, a reactor scram occurs, the discharge of radioactive secondary system coolant would occur as steam through the seconday coolant system (steam generator) relief valves.

2. POSTULATED CAUSES

Because the steam generator tube material is ductile, the assumption of a complete severance is considered conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Steam generator tube denting, resulting in minor leaks has been observed in a number of operating PWRs, and complete severance failure due to the pressure differential across the tube (greater than 100 psi) at a weakened denting location is a possibility. Radioactivity in the secondary coolant system is subject to continuous monitoring, and the cumulative effects of minor leaks from the primary coolant system to the secondary coolant system are not permitted to exceed Technical Specification limits.

3. ACCIDENT DESCRIPTION

The accident evaluation assumes the complete severance of a single steam generator tube. The accident is assumed to take place at full power with the primary coolant contaminated with fission products corresponding to continuous operation with 0.5% defective fuel. The accident is further assumed to lead to an increase in contamination of the secondary coolant system due to leakage of primary coolant from the primary to the secondary coolant system. In the event of a coincident loss of offsite power, or failure of the condenser dump (steam bypass) system, discharge of activity to the atmosphere would take place via the steam generator safety or relief valves.

The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit[a]. The recovery procedure can be carried out on a time scale which ensures that the break flow to the secondary coolant system is terminated before the water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily[6].

The rate of release of radioactivity following a steam generator tube rupture depends on the primary-to-secondary coolant leakage rate, the percentage of defective fuel in the core, and the duration of blowdown resulting from the rupture.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The maximum point of airborne fission product concentration following a steam generator tube rupture can occur at the steam relief valve discharge if an accompanying loss of power/loss of air ejector occurs, or at the condenser vacuum system discharge (noncondensibles only). Both of these locations are usually above the elevation of the turbine building roof.

4.1.2 Principal Point of Environmental Release. For the steam generator tube rupture accident, the principal points of environmental release are the condenser vacuum system discharge vent and the steam relief valves. This evaluation considers only the release from the condenser vacuum system vent. A release from the steam relief valves would contain somewhat larger quantities of iodine due to greater carryover of halogens.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case

The assumptions and accident conditions for this case are taken from NUREG-0099, Appendix I[4] or from SARs for PWR plants currently under NRC review.

The assumptions used to determine the equilibrium concentrations of radionuclides in the secondary system prior to the accident are as follows:

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[a] In some plants, isolation of a steam generator is not possible and the reactor must be shut down.

(1) The reactor is operated at an assumed core thermal power of 3800 MWt for a period sufficient to establish equilibrium concentrations of the radioactive nuclides in the primary coolant (620 days) with 0.5% defective fuel.

(2) A 100 lb/day primary-to-secondary coolant leakage rate of sufficient duration to result in equilibrium steam system fission product concentration is assumed to exist during plant operation[4].

(3) Fifteen percent of the average inventory of noble gases and halogens in the primary coolant is assumed to be released into the secondary coolant[4].

(4) All of the noble gas activity plus 0.1% of the halogen activity released to the secondary coolant system from the primary coolant system as a direct result of the accident plus the equilibrium activity in the secondary system due to an assumed 100 lb/day primary-to-secondary coolant leakage is released over a 30-minute period. Decay is taken into account. (Contribution to the atmospheric release from the equilibrium secondary coolant activity is insignificant compared with that released to the secondary coolant system from the primary coolant system during the accident.)

(5) Only one steam generator is affected by the severance of a steam generator tube.

(6) Thirty minutes after the accident, the pressure between the defective steam generator and the primary coolant system is equalized. The defective unit is isolated. No steam and fission product activities are released from the defective steam generator thereafter.

Table XIX (Maximum Case) presents the calculated maximum radionuclide activities released to the atmosphere from the condenser vacuum system following a steam generator tube rupture.

5.2 Case 2 – Realistic Case

The only assumption differing from those for the maximum case is that 0.12% defective fuel, instead of 0.5%, is assumed[5].

Table XX (Realistic Case) presents the calculated radionuclide activities released to the atmosphere from the condenser vacuum system following a steam generator tube rupture.

TABLE XIX

RADIOACTIVITY RELEASED FROM STEAM GENERATOR TUBE RUPTURE
(Maximum Case)

Radionuclide	Radioactivity Released to Atmosphere (Ci)
Br-83	7.0(-4)[a]
Br-84	3.0(-4)
Br-85	6.5(-6)
I-130	3.2(-4)
I-131	4.2(-2)
I-132	1.4(-2)
I-133	5.9(-2)
I-134	6.1(-3)
I-135	2.9(-2)
Kr-83m	3.0
Kr-85m	1.7
Kr-85	2.3
Kr-87	8.2
Kr-88	2.9(1)
Kr-89	1.2(-1)
Xe-131m	1.7(1)
Xe-133m	3.4(1)
Xe-133	2.8(3)
Xe-135m	2.0
Xe-135	5.4(1)
Xe-137	2.6(-1)
Xe-138	3.6
Total	3.0(3)

[a] Example: 7.0(-4) = 7.0×10^{-4}.

TABLE XX

RADIOACTIVITY RELEASED FROM STEAM GENERATOR TUBE RUPTURE
(Realistic Case)

Radionuclide	Radioactivity Released to Atmosphere (Ci)
Br-83	1.7(-4)[a]
Br-84	7.2(-5)
Br-85	1.5(-6)
I-130	7.8(-5)
I-131	1.0(-2)
I-132	3.5(-3)
I-133	1.4(-2)
I-134	1.5(-3)
I-135	6.9(-3)
Kr-83m	7.2(-1)
Kr-85m	4.0(-1)
Kr-85	5.6
Kr-87	2.0
Kr-88	7.0
Kr-89	2.8(-2)
Xe-131m	4.1
Xe-133m	8.2
Xe-133	6.7(2)
Xe-135m	4.8(-1)
Xe-135	1.3(1)
Xe-137	6.2(-2)
Xe-138	8.7(-1)
Total	7.1(2)

[a] Example: $1.7(-4) = 1.7 \times 10^{-4}$.

VIII. ACCIDENT CLASS 6.0 – REFUELING ACCIDENTS
(In Containment)

Accident 6.1 – Fuel Bundle Drop
(In Containment)

Accident 6.2 – Heavy Object Drop Onto Fuel in Core

1. INTRODUCTION

The possibility exists for the release of fission products from the fuel as a result of cladding damage at any time irradiated fuel is being handled. While fuel is being handled within the containment building, any release of fission products from an accident would be confined principally to the containment building. The release of fission products warrants evaluation of radiological consequences of a fuel handling accident within the containment building.

2. POSTULATED CAUSES

After the reactor head and rod cluster control drive shafts are removed, fuel bundles are lifted individually from the core, transferred to the refueling pool, placed horizontally on a conveyor car and pulled through the transfer tube and canal, upended and transferred through the spent fuel pool gate, then lowered into steel racks for storage in the spent fuel pool.

The containment building, refueling pool, transfer tube and canal, and the spent fuel pool are designed in accordance with NRC guidelines[8], which prevent the structures from failing in the event of an earthquake. They are also designed to prevent any credible external missile from entering the buildings and to prevent any internal missile from penetrating the walls of these structures. The fuel handling manipulators, cranes, trollies, bridges, and associated equipment, above the water cavities through which the fuel bundles move, are designed to prevent this equipment from generating missiles and damaging the fuel.

Movement of fuel handling equipment is kept at low speeds while caution is observed that the fuel does not strike another object during transfer from the core to its storage position.

The design of the fuel bundle is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 60 pounds on each fuel rod. Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, damage to the individual fuel rods during the normal course of handling is unlikely. If one bundle were lowered on top of another, no damage to the fuel rods would occur that would breach the cladding.

If, during handling, the fuel bundle strikes a flat surface, the loads would be distributed across the fuel bundles and grid clips, and essentially no damage would be expected in any fuel rods. If the fuel bundle were to strike a sharp object, fuel rod damage might occur, but breach of the cladding would not be expected.

Assumptions are made for two postulated accidents: one in which an entire fuel bundle (264 fuel rods) is damaged and the other where only an outer row of fuel rods of a single bundle is damaged (17 fuel rods).

3. ACCIDENT DESCRIPTION

During the fuel handling operations, the containment building is kept in an isolable condition with all penetrations to the outside atmosphere either closed or capable of being closed on an alarm signal from a monitor indicating that radioactivity is above prescribed limits within the containment building.

Should a fuel bundle be damaged and release activity above a prescribed level, radioactivity monitors within the containment building would sound an audible alarm; the containment building would be isolated, and personnel would be evacuated.

During refueling, the containment building ventilation system is assumed to exhaust about 15,000 cfm of air from the surface of the refueling pool through the refueling pool exhaust subsystem and about 35,000 cfm of air from the general containment building atmosphere. This total of approximately 50,000 cfm is further assumed to be exhausted through the containment ventilation isolation valves, HEPA filters, and charcoal adsorbers.

3.1.1 Sequence of Containment Building Isolation Following a Postulated Refueling Accident. Following the postulated refueling accident, airborne radioactivity is released from the surface of the refueling pool where it mixes with air above the pool and is exhausted by the refueling pool exhaust system. In this evaluation, it was assumed that the leading edge of airborne radioactivity in the exhaust duct passes the isolation valves approximately 15 seconds before reaching the monitor probe. The monitor line transit time from the probe to the detector is approximately 3 seconds. Since for the assumed accident the initial concentration of airborne radioactive materials will be substantially higher than twice background, the "high alarm" contacts in the radioactivity monitors would close in approximately 0.5 second. A containment building ventilation isolation signal would be generated in approximately 0.02 second with closure of the purge exhaust isolation valves being initiated approximately 0.5 second after receipt of the signal. The inboard motor-operated purge exhaust valve closure time is about 5 seconds. Under the assumed condition, and sequence of operation, the isolation valves would be closed about 23 seconds after the gaseous radioactivity enters the exhaust duct system.

3.1.2 Radioactivity Exhausted by Refueling Pool Exhaust System and Released to Atmosphere. The refueling pool is rectangular with approximate dimensions of 70 x 20 feet. The combination of the rapidly decreasing air velocity away from the exhausters at a relatively short distance from the edge of the pool, the thermal convection from the warmer pool surface, and the turbulence induced by the pool supply air and the containment air coolers, will result in the rapid mixing of the radioactive gases evolving from the pool within the containment building atmosphere. It was assumed that the evolved airborne radioactivity was retained within the (approximately 20,000 ft^3) volume formed by the pool surface and the missile barrier. Where the missile barrier does not surround the pool, the radioactivity would actually be dispersed into a larger volume of air which would have the effect of reducing the concentrations. For conservatism, however, it was assumed that all the radioactivity remained within this 20,000 ft^3 volume.

3.1.3 Action Following Containment Isolation. Following containment building isolation after the fuel handling accident, the radioactivity can be removed from the containment building atmosphere by cleanup recirculation units which consist of HEPA filters and charcoal adsorbers in series, with a typical capacity of about 10,000 cfm per train. Most plants have two redundant cleanup recirculation units.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The position above the refueling pool directly over the fuel bundle postulated to be damaged is the point of maximum concentration. With the various air flow directions and thermal convection, however, practically any point in the volume bounded by the pool surface and the missile barrier may be considered the location of maximum concentration.

4.1.2 Principal Point of Environmental Release. The containment building ventilation purge system is assumed to exhaust 50,000 cfm through the containment isolation valves and HEPA filters, and out the exhaust duct near the top of the containment building until the isolation valves are closed.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case

This case, which approximates Accident 6.2, NUREG-0099 (Appendix I)[4], makes use of typical plant parameters (such as maximum allowable isolation valve closure times) and some of the assumptions of Regulatory Guide 1.25[7].

The assumptions relating to the release of radioactive material from the fuel as a result of the postulated refueling accident are as follows:

(1) The accident occurs 100 hours after shutdown, the earliest time fuel handling operations may begin. Radioactive decay of the fission products was taken into consideration during this time period[4].

(2) All fuel rods in one bundle (264) are assumed to be damaged as a result of the handling accident[4].

(3) All of the gap activity in the damaged rods is released to the refueling pool water and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioiodine in the rods[7].

(4) Fuel rod fission product inventories are based on full power operation at the end of core life immediately preceding shutdown and a radial peaking factor of 1.65[7].

(5) The pool decontamination factor for radioiodine is 500[4].

(6) The retention of noble gases in the refueling pool is negligible (i.e., decontamination factor of 1)[7].

(7) In that a significant number of events occur which affect concentrations within the containment building, the calculations assume: (a) instantaneous release of the radioactivity into a 20,000 ft^3 volume, (b) discharge of the mixture at a rate of 15,000 cfm for 23 seconds, and (c) after containment building isolation, mixing of the remaining radioactivity within the assumed containment building volume of 2×10^6 ft^3.

Table XXI shows the calculated radionuclide releases from the maximum postulated fuel handling accident and the resulting concentrations above the refueling pool and within the containment building.

5.2 Case 2 – Realistic Case

This case approximates Accident 6.1 of NUREG-0099 (Appendix I)[4].

The assumptions of the maximum release case are modified for the realistic case as follows:

(1) The activity release is 17/264 that of the maximum release case (17 rods instead of 264)[3]

TABLE XXI

RADIOACTIVITY RELEASED FROM POSTULATED FUEL HANDLING ACCIDENT DURING REFUELING (Maximum Case)

Radionuclide	Radioactivity Released Above Pool (Ci)	Peak Radioactivity Concentration (μCi/cc)	
		Above Pool	Containment Building
I-131	1.3(2)[a]	2.3(-1)	1.7(-3)
I-132	1.1(2)	1.9(-1)	1.4(-3)
I-133	1.4(1)	2.4(-2)	1.8(-4)
I-135	1.0(-2)	1.8(-5)	1.4(-7)
Kr-85	2.3(3)	4.0	3.0(-2)
Xe-131m	7.1(2)	1.2	9.4(-3)
Xe-133m	1.8(3)	3.3	2.4(-2)
Xe-133	1.2(5)	2.2(2)	1.6
Xe-135m	1.6	2.8(-3)	2.1(-5)
Xe-135	2.5(2)	4.3(-1)	3.3(-3)
Total	1.3(5)	2.3(2)	1.7

[a] Example: $1.3(2) = 1.3 \times 10^2$.

(2) Discharge of the mixture is at a rate of 15,000 cfm for 20 seconds.

(3) One week decay time is assumed before the accident occurs[4].

Table XXII presents the calculated radionuclide releases from the postulated "realistic" accident and the resulting concentrations above the refueling pool and within the containment building.

TABLE XXII

RADIOACTIVITY RELEASED FROM POSTULATED FUEL HANDLING ACCIDENT DURING REFUELING (Realistic Case)

Radionuclide	Radioactivity Released Above Pool (Ci)	Peak Radioactivity Concentration (μCi/cc)	
		Above Pool	Containment Building
I-131	6.6	1.2(-2)[a]	8.8(-5)
I-132	3.8	6.7(-3)	5.1(-5)
I-133	9.3(-2)	1.6(-4)	1.2(-6)
Kr-85	1.5(2)	2.6(-1)	1.9(-3)
Xe-131m	4.3(1)	7.6(-2)	5.7(-4)
Xe-133m	5.3(1)	9.3(-2)	7.0(-4)
Xe-133	5.6(3)	9.8	7.4(-2)
Xe-135	1.0(-1)	1.8(-4)	1.3(-6)
Total	5.9(3)	1.0(1)	7.7(-2)

[a] Example: 1.2(-2) = 1.2×10^{-2}.

IX. ACCIDENT CLASS 7.0 – SPENT FUEL HANDLING ACCIDENTS

OUTSIDE CONTAINMENT

Accident 7.1 – Fuel Assembly Drop in Fuel Storage Pool

Accident 7.2 – Heavy Object Drop Onto Fuel Rack

1. INTRODUCTION

Fuel handling accidents inside the containment building were discussed under Accident Class 6. From the containment building, spent fuel is moved to a fuel storage pool, usually located in an adjacent building, pending shipment to an offsite location for reprocessing, storage, or disposal. During movement of spent fuel, or in the movement of heavy objects near spent fuel, the possibility exists for damage to spent fuel and for release of fission products from damaged fuel.

2. POSTULATED CAUSES

After the fuel bundles have been moved from the containment building to the spent fuel pool building (or fuel storage facility), they are upended and transferred through a spent fuel pool gate, then lowered into racks for storage in the spent fuel storage pool in a pattern which prevents any possibility of a criticality accident.

The spent fuel storage pool is designed in accordance with NRC guidelines[8], which prevents failure in the event of an earthquake. It is also designed to prevent any credible external missile from entering the buildings and reaching the stored irradiated fuel, and any internal missile from penetrating the walls of these structures. The fuel handling manipulators, cranes, trollies, bridges, and associated equipment are designed to prevent this equipment from generating missiles and damaging the fuel. The construction of the fuel bundles precludes damage to the fuel should small objects, such as portable or hand tools, drop onto a bundle.

The facility is designed so that heavy objects, such as the fuel cask, cannot be moved over the irradiated fuel stored in the spent fuel storage pool, and only one fuel bundle can be handled at a time. Movement of fuel handling equipment is maintained at low speeds while caution is taken that the fuel does not strike other objects or structures during transfer from the core to storage positions.

If one bundle is lowered on top of another, no damage to the fuel rods is expected. If, during handling, the fuel bundle strikes against a flat surface, the loads would be distributed across the fuel bundles and grid clips, and no damage would be expected in any fuel rods. If the fuel bundle were to strike a sharp object, it is possible that the sharp object might damage the fuel rods it comes in contact with, but cladding breach is unlikely.

Either the failure of equipment to function as designed or operator error can be postulated to cause fuel cladding failure. For example, moving the bridge crane with an element only partially within a storage rack or dropping a fuel bundle would probably result in release of fission products from damaged fuel rods.

3. ACCIDENT DESCRIPTION

A fuel bundle conceivably could be damaged in the transfer tube and canal or in the spent fuel storage pool. However, supply air for the spent fuel storage pool area is swept across the fuel pool and transfer canal, and exhausted through the plant vent. An area radiation monitor is usually located on the bridge over the spent fuel storage pool, and portable radiation monitors with audible alarms are located in the area during fuel handling operations. Doors in the spent fuel storage pool area are closed to maintain controlled leakage characteristics in the spent fuel storage pool region during refueling operations involving irradiated fuel. Should a fuel bundle be dropped in the canal or in the pool and release radioactivity above a prescribed level, radioactivity monitors would alarm and the spent fuel storage pool ventilation air would be exhausted through charcoal adsorbers and HEPA filters to remove most of the halogens prior to discharging it to atmosphere.

If the discharge vent radiation monitor should indicate that the radioactivity in the vent discharge is above prescribed levels, a radioactivity monitor would alarm, and the supply and exhaust ventilation systems servicing the spent fuel pool building could be shut down to limit leakage to the atmosphere.

Any movement of the fuel cask in the spent fuel pool building is under administrative control. Interlocks prevent the crane from moving the cask over stored irradiated fuel and also limit cask movement.

Shock-absorbing analyses indicate that in most incidents where a fuel bundle strikes another object, the outer row of fuel rods experiences greater loads and stresses than the inner rows. Therefore, if a fuel bundle drops, not necessarily all the fuel rods break. For the fuel handling accident analysis, two accidents are considered: one in which the cladding of all the fuel rods in one fuel bundle break suddenly, releasing all the gaseous fission products in the voids between the fuel pellets; and the other in which the outer row of fuel rods is damaged.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The position in the spent fuel storage pool directly above the fuel bundle postulated to be damaged is the point at which a maximum concentration of airborne radionuclides would occur. In general, the position can be best assumed as that of the crane handling the fuel or objects above the fuel.

4.1.2 Principal Point of Environmental Release. As discussed under the accident description, the exhausted air from the spent fuel pool building is assumed to be passed through HEPA filters, charcoal adsorbers, and a second set of HEPA filters in series. A ventilation system is assumed to sweep approximately 20,000 cfm from the spent fuel pool building which would be mixed with exhaust air from other buildings or facilities and subsequently discharged from a vent near the top of the containment building.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case

This case, which approximates Accident 7.2, NUREG-0099 (Appendix I)[4], utilizes parameters for the worst postulated fuel handling accidents. The decay times of Accidents 7.1 and 7.2 were interchanged to maximize the releases from the more severe accident. Several of the assumptions are derived from Regulatory Guide 1.25[7].

(1) The accident is assumed to occur one week after shutdown[4]. Radioactive decay of the fission products was taken into consideration during this time period.

(2) All fuel rods in one bundle (264) are assumed to be damaged as a result of the accident[4].

(3) All of the gap activity in the damaged rods is assumed to be released to the pool water and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioiodine in the rods[7]. Particulates are assumed to remain in the fuel rods or to be retained by the pool water.

(4) Bundle fission product inventories are based on full-power operation at the end of core life immediately preceding shutdown and on a radial peaking factor of 1.65[7].

(5) The pool decontamination factor for the radioiodines is assumed to be 500[4].

(6) Noble gas decontamination factor is 1.0 for the pool[7].

The analysis of the activity releases from the spent fuel storage pool water resulting from the postulated fuel handling accident is based on several of the fission product source and release assumptions of Regulatory Guide 1.25. This conservative approach to the evaluation of radiological consequences assumes that the bundle with the peak fission product inventory is the one damaged. The inventory was calculated assuming maximum full-power operation at the end of core life immediately preceding shutdown and includes a radial peaking factor.

Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation is assumed to be available for immediate release to the water following clad damage. Compared to the quantity immediately released, all subsequent activity releases are considered to be negligible.

Table XXIII shows the calculated radionuclide released from the postulated maximum spent fuel storage pool accident and the resulting concentrations above the pool and in the exhaust system.

TABLE XXIII

RADIOACTIVITY RELEASED FROM POSTULATED FUEL HANDLING ACCIDENT IN SPENT FUEL STORAGE POOL (Maximum Case)

Radionuclide	Radioactivity Released (Ci)	Peak Radioactivity Concentration (μCi/cc)	
		Above Pool	Exhaust System
I-131	1.3(2)[a]	4.0	8.0(-3)
I-132	1.1(2)	3.3	6.6(-3)
I-133	1.4(1)	4.1(-1)	8.3(-4)
I-135	1.0(-2)	3.1(-4)	6.2(-7)
Kr-85	2.3(3)	7.0(1)	1.4(1)
Xe-131m	7.1(2)	2.2(1)	4.3
Xe-133m	1.9(3)	5.7(1)	1.1(1)
Xe-133	1.2(5)	3.7(3)	7.5(2)
Xe-135m	1.6	4.9(-2)	9.7(-3)
Xe-135	2.5(2)	7.5	1.5
Total	1.3(5)	3.9(3)	7.8(2)

[a] Example: 1.3(2) = 1.3×10^2.

5.2 Case 2 – Realistic Case (Accident 7.1)

The assumptions related to a more probable release of radioactive material from the spent fuel pool building as a result of the postulated fuel handling accident which differ from the maximum release case are as follows:

(1) The accident is assumed to occur 30 days after shutdown[4]

(2) Seventeen fuel rods are assumed to be damaged as a result of the handling accident[4].

Table XXIV shows the calculated radionuclide releases from a postulated "realistic" spent fuel pool building accident and the resulting concentrations above the pool and in the exhaust system.

TABLE XXIV

RADIOACTIVITY RELEASED FROM POSTULATED FUEL HANDLING ACCIDENT IN SPENT FUEL STORAGE POOL (Realistic Case)

Radionuclide	Radioactivity Released (Ci)	Peak Radioactivity Concentration (μCi/cc) Above Pool	Exhaust System
I-131	9.2(-1)[a]	2.8(-2)	5.6(-5)
I-132	2.8(-2)	8.6(-4)	1.7(-6)
Kr-85	4.9(1)	1.5	3.0(-1)
Xe-131m	1.8(1)	5.5(-1)	1.1(-1)
Xe-133m	4.6(-2)	1.4(-3)	2.8(-4)
Xe-133	2.7(2)	8.3	1.7
Total	3.4(2)	1.0(1)	2.1

[a] Example: 9.2(-1) = 9.2×10^{-1}.

X-A. ACCIDENT CLASS 8.0 – ACCIDENT INITIATION EVENTS CONSIDERED IN DESIGN BASIS EVALUATION IN THE SAFETY ANALYSIS REPORT

Accident 8.1 – Loss-of-Coolant Accidents

1. INTRODUCTION

A loss-of-coolant accident (LOCA) would result from rupture of the primary coolant system pressure boundary. This pressure boundary includes all primary coolant system piping, components, and connecting lines up to and including the first closed isolation valve. Ruptures of small lines or failures of seals such as those in pumps, valves, and bolted closures result in small cross-sectional flow areas, and the plant charging or makeup systems can usually maintain primary system pressure and water inventory to allow an orderly shutdown. Should a larger flow area be opened by component or pipe rupture, the fuel cladding could fail due to the severe thermal transient, and release of fission products to the containment building atmosphere would occur.

2. POSTULATED CAUSES

The failure of any part of the primary coolant system pressure boundary which could subsequently cause significant fuel damage is unlikely. Failure of some pressure boundary components, such as valve packing, pump seals, and flange seals is probable during the life of the plant, and the radioactivity within the primary coolant can be dispersed into the containment building. In order to bound the radiological release concentrations of a LOCA, two cases are considered: one for postulated rupture of a large pipe and the other for the equivalent of a small pipe break.

3. ACCIDENT DESCRIPTION

A LOCA could result from a rupture of the primary coolant system or of any line connected to that system up to the first closed valve. A rupture resulting in a very small cross-sectional flow area would cause a leak of primary coolant at a rate which could be accommodated by the charging or makeup pumps. Should such a small rupture occur, these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown.

Should a larger break occur, resultant loss of pressure and pressurizer liquid level or high containment building air pressure would initiate a reactor scram and activate the emergency core cooling system (ECCS).

The ECCS would limit the cladding temperature and reduce the potential for, or extent of, metal-water reactions. A large leak of primary coolant would flash to steam, would cause an increase in containment building air pressure, and would distribute fission products throughout the containment building. The containment building isolation system would be activated and would limit escape of fission products to those which were discharged prior to isolation and those which leak from the containment building due to its pressurization at a low rate. Further release of fission products outside the containment building can result (through leaking valves and pump seals) when recirculation of the ECCS fluids and of the spilled primary coolant begins, along with operation of the residual heat removal/low pressure injection system.

At the time a LOCA occurs, the containment spray system may be activated for the reduction of airborne fission products (principally iodines) within the containment building. As this system also recirculates contaminated water external to the containment building, leakage from bonnets, valve seats, and pump seals of this system is also a source for airborne fission products. Where available, the operation of air cleanup recirculation units with HEPA filters and charcoal adsorbers would reduce the concentration of radioiodines available for leakage from the containment building.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The maximum concentration of airborne fission products would occur within the containment building soon after initiation of a LOCA; however, the severe steam/water environment makes it difficult to design reliable radiation monitoring instrumentation for use in containment.

4.1.2 Principal Points of Environmental Release. Release of radioactive materials to the environment during and immediately following a LOCA can occur by several paths. One such path is release of a quantity of radioactive primary coolant system liquid or vapor through the containment building ventilation system until the system is isolated. A second pathway is leakage from the containment building which can occur at penetrations, such as airlocks, piping penetrations connecting to the primary coolant or containment atmosphere, and electrical penetrations; such leaks, however, are expected to have very low flow rates. A third path is leakage from the external fluid recirculation systems. Each of the three paths described have many potential release points.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case (Large Pipe Break)

The assumptions and conditions for the large pipe break accident are in accordance with the general assumptions of NUREG-0099 (Regulatory Guide 4.2, Rev. 1) Appendix I[4].

(1) The primary coolant activity is based on 0.5% failed fuel, plus the release of 2% of the core inventory of halogens and noble gases as a result of cladding failure during the LOCA[4]. The cladding failure is assumed to occur after containment building isolation. The reactor is assumed to have operated at 3800 MWt for 620 days prior to the accident.

(2) All of the primary coolant is assumed to be released to the containment building.

(3) A halogen reduction factor of 0.5 is applied to the primary coolant source[4].

(4) Containment building volume is assumed to be approximately 2×10^6 ft^3, and 70% building mixing is assumed[5].

Table XXV presents the calculated radionuclide activities released and peak concentrations within the containment building resulting from a postulated large primary coolant pipe rupture.

5.2 Case 2 – Less Severe Case (Small Pipe Break)

The assumptions differing from the large pipe break case are as follows:

(1) No additional fuel failure occurs as a result of the LOCA[4].

(2) The activity in the primary coolant is based on 0.12% failed fuel[5].

Table XXVI presents the calculated radionuclide activities released and peak containment building concentrations resulting from the rupture of a small primary coolant pipe.

TABLE XXV

RADIOACTIVITY RELEASED FROM LARGE PRIMARY COOLANT PIPE RUPTURE

Radionuclide	Radioactivity Released (Ci)	Peak Containment Building Airborne Concentration (μCi/cc)
Br-83	9.6(4)[a]	2.4
Br-84	2.2(5)	5.5
Br-85	2.3(5)	5.7
I-130	1.3(4)	3.4(-1)
I-131	1.1(6)	2.7(1)
I-132	1.5(6)	3.9(1)
I-133	2.1(6)	5.2(1)
I-134	2.4(6)	6.0(1)
I-135	1.9(6)	4.9(1)
Kr-83m	1.9(5)	4.9
Kr-85m	5.6(5)	1.4(1)
Kr-85	1.9(4)	4.9(-1)
Kr-87	1.1(6)	2.7(1)
Kr-88	1.6(6)	4.0(1)
Kr-89	1.6(6)	4.1(1)
Xe-131m	1.8(4)	4.6(-1)
Xe-133m	1.0(5)	2.6
Xe-133	4.3(6)	1.1(2)
Xe-135m	1.1(6)	2.8(1)
Xe-135	1.4(6)	3.5(1)
Xe-137	3.6(6)	9.2(1)
Xe-138	3.7(6)	9.4(1)
Total	2.9(7)	7.3(2)

[a] Example: $9.6(4) = 9.6 \times 10^4$.

TABLE XXVI

RADIOACTIVITY RELEASED FROM SMALL PRIMARY COOLANT PIPE BREAK

Radionuclide	Radioactivity Released (Ci)	Peak Containment Building Airborne Concentration (μCi/cc)
Kr-83m	5.0	1.3(-4)[a]
Kr-85m	2.6(1)	6.6(-4)
Kr-85	3.6(1)	9.0(-4)
Kr-87	1.4(1)	3.6(-4)
Kr-88	4.8(1)	1.2(-3)
Kr-89	1.2	3.0(-5)
Xe-131m	2.6(1)	6.6(-4)
Xe-133m	5.3(1)	1.3(-3)
Xe-133	4.3(3)	1.1(-1)
Xe-135m	3.1	7.8(-5)
Xe-135	8.4(1)	2.1(-3)
Xe-137	2.2	5.4(-5)
Xe-138	1.1(1)	2.7(-4)
Br-83	5.8(-1)	1.5(-5)
Br-84	3.1(-1)	7.9(-6)
Br-85	3.6(-2)	9.1(-7)
I-130	2.5(-1)	6.4(-6)
I-131	3.2(1)	8.2(-4)
I-132	1.2(1)	3.0(-4)
I-133	4.6(1)	1.1(-3)
I-134	5.6	1.4(-4)
I-135	2.3(1)	5.7(-4)
H-3	2(2)[b]	6(-3)
N-16	1(4)	2(-1)
Cr-51	5(-1)	1(-5)
Mn-54	7(-2)	2(-6)
Fe-55	4(-1)	1(-5)
Fe-59	2(-1)	6(-6)
Co-58	4	1(-6)
Co-60	5(-1)	1(-5)
Rb-86	2(-2)	5(-7)
Rb-88	5(1)	1(-3)
Sr-89	8(-2)	2(-6)
Sr-90	2(-3)	6(-8)
Sr-91	2(-1)	4(-6)
Y-90	3(-4)	7(-5)
Y-91m	9(-2)	2(-6)
Y-91	2(-2)	4(-7)
Y-93	8(-3)	2(-7)

TABLE XXVI (continued)

Radionuclide	Radioactivity Released (Ci)	Peak Containment Building Airborne Concentration (μCi/cc)
Zr-95	1(-2)	4(-7)
Nb-95	1(-2)	3(-7)
Mo-99	2(1)	5(-4)
Tc-99m	1(1)	3(-4)
Ru-103	1(-2)	3(-7)
Ru-106	2(-3)	6(-8)
Rh-103m	1(-2)	3(-7)
Rh-106	2(-3)	6(-8)
Te-125m	7(-3)	2(-7)
Te-127m	7(-2)	2(-6)
Te-127	2(-1)	5(-6)
Te-129m	3(-1)	8(-6)
Te-129	4(-1)	1(-5)
Te-131m	6(-1)	2(-5)
Te-131	3	7(-5)
Te-132	7	2(-4)
Cs-134	6	2(-4)
Cs-136	3	8(-5)
Cs-137	4	1(-4)
Ba-137m	4	1(-4)
Ba-140	5(-2)	1(-6)
La-140	4(-2)	9(-7)
Ce-141	2(-2)	4(-7)
Ce-143	1(-2)	2(-7)
Ce-144	8(-3)	2(-7)
Pr-143	1(-2)	3(-7)
Pr-144	8(-3)	2(-7)
Np-239	3(-1)	7(-6)
Total	2(4)	4(-1)

[a] Example: 1.3(-4) = 1.3×10^{-4}.

[b] Data accuracy to one significant digit.

X-B. ACCIDENT CLASS 8.0 – ACCIDENT INITIATION EVENTS CONSIDERED IN DESIGN BASIS EVALUATION IN THE SAFETY ANALYSIS REPORT

Accident 8.2(a) – Rod Ejection Accident

1. INTRODUCTION

The rupture of a control rod drive mechanism housing and associated rod cluster control assembly ejection can result in an accident with a rapid reactivity insertion together with an adverse core power distribution as well as a loss of coolant through the pressure boundary breach.

2. POSTULATED CAUSES

Certain design features in pressurized water reactors are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. Even though conservative design, analysis, and testing principles are employed, the failure of welds due to stress, corrosion, or undetected defects is possible, and a rod ejection accident conceivably could occur.

3. ACCIDENT DESCRIPTION

The rupture of a control rod drive mechanism housing due to mechanical damage, weld failure, or other failure in the pressure boundary can result in a large differential pressure being applied across parts of the rod cluster control assembly, resulting in ejection of the rod from its normal position in the reactor core. The ejection of the rod would cause a rapid reactivity insertion with adverse core power distribution. This would further cause some fuel rods to experience departure from nucleate boiling (DNB) resulting in cladding failures. A breach in the primary coolant pressure boundary is basically a loss-of-coolant accident (see Accident 8.1) and the resulting accident evaluation is similar, with the exception that the power excursion can cause sufficiently high secondary system pressure to lift the secondary system relief valves.

Loss of pressure and pressurizer liquid level would initiate reactor scram, actuation of the ECCS, and containment building isolation.

Leakage of primary coolant would cause an increase in containment building pressure and would distribute fission products throughout the containment building. The containment building isolation system is assumed to activate and would limit the escape of fission products to those which were discharged prior to isolation and to subsequent slow leakage from the containment (due to its pressurization). Further release of fission products outside the containment building can result through leaking valves and pump seals when recirculation of the spilled primary coolant and ECCS fluids begins, along with operation of the residual heat removal/low pressure injection system.

At the time the rod injection occurs, the containment spray system may be activated for the reduction of airborne fission products (principally iodines) within the containment building. Since this system also recirculates contaminated water external to the containment building, leakage from bonnets, valve seats, and pump seals are also potential sources of airborne fission products in the auxiliary building. The operation of air cleanup recirculation units with HEPA filters and charcoal adsorbers would reduce the concentration of radioiodines available for leakage from the containment building.

Prior to the accident, it is assumed that the plant had been operating with simultaneous fuel defects and steam generator tube leakage for sufficient time to establish equilibrium levels of activity in the primary and secondary cooling systems.

The activity available for release to the atmosphere from the relief valves would be the equilibrium activity in the secondary coolant plus that fraction of the activity leaking from the primary coolant through the steam generator tubes. The leakage of primary coolant to the secondary side of the steam generator is assumed to continue until the pressures in the primary coolant and secondary coolant systems equalize. No mass transfer from the primary coolant system to the secondary coolant system due to the steam generator tube leakage would occur thereafter.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The maximum concentration of airborne fission products would occur within the containment building soon after rod ejection.

4.1.2 Principal Points of Environmental Release. Environmental release resulting from a rod ejection accident can occur by several pathways. The principal and most probable pathway would be release of a quantity of primary coolant system activity as airborne gases and particulates through the containment building ventilation system until the containment is isolated. The second pathway would be release through containment building leakage, most of which would occur at low flow rates through penetrations such as airlocks, piping penetrations connecting to the primary coolant or containment building atmosphere, and electrical penetrations. A third pathway would be leakage from the external fluid recirculation systems, and a fourth would be through the secondary coolant system relief valves.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case

The assumptions and accident conditions are in accordance with the general assumptions of NUREG-0099 (Regulatory Guide 4.2, Rev. 2) Appendix I[4].

(1) The primary coolant activity is based on 0.5% failed fuel, plus release of 0.2% of the core inventory of halogens and noble gases as a result of cladding failure during the rod ejection. Cladding failure is assumed to occur after containment building isolation. The reactor is assumed to have operated at 3800 MWt for 620 days.

(2) Containment building volume is assumed to be 2×10^6 ft^3, and 70% building mixing is assumed.

(3) A continuous steam generator blowdown rate of 10 gpm is assumed[4] but is insignificant.

(4) The release of primary coolant to the containment building is assumed to occur instantaneously. The total release of primary coolant into the containment building is assumed to be 3×10^5 pounds.

(5) The primary-to-secondary coolant leakage rate is assumed to be constant at 100 lb/day (the leak rate prior to the accident) until the pressure of the two systems equalize.

(6) Primary coolant and secondary coolant system pressures are assumed to be equalized in approximately 20 minutes, terminating primary-to-secondary coolant leakage in the steam generators.

(7) Steam dump to the atmosphere is assumed to occur during the first four minutes after the rod ejection. During this period, a total of about 60,000 pounds is assumed to be vented to the atmosphere.

(8) All of the noble gases which leak into the secondary coolant system after the accident are assumed to be discharged to the atmosphere during the first two hours.

(9) After the accident the partition factor of iodine is conservatively assumed to be 0.1 for U-tube steam generators and 1.0 for once-through steam generators.

Table XXVII presents a listing of the radionuclides released and their calculated activities resulting from the steam dump from the secondary coolant system to the atmosphere mentioned in assumption (7) above. Release of radioactivity was found to be independent of the type of steam generator plant (U-tube versus once-through).

TABLE XXVII

RADIOACTIVITY IN STEAM RELEASE TO ENVIRONMENT
FROM ROD EJECTION ACCIDENT
(Maximum Case)

Radionuclide	Radioactivity Released (Ci)
Kr-83m	3.5(-1)[a]
Kr-85m	1.2
Kr-85	4.7(-2)
Kr-87	1.7
Kr-88	3.2
Kr-89	2.7(-1)
Xe-131m	4.4(-2)
Xe-133m	2.6(-1)
Xe-133	1.1(1)
Xe-135m	2.6
Xe-135	3.3
Xe-137	6.6(-1)
Xe-138	1.8
Br-83	6.5(-4)
Br-84	1.5(-3)
Br-85	1.5(-3)
I-130	9.3(-5)
I-131	8.5(-3)
I-132	1.1(-2)
I-133	1.5(-2)
I-134	1.6(-2)
I-135	1.3(-2)
Total	2.6(1)

[a] Example: $3.5(-1) = 3.5 \times 10^{-1}$

Table XXVIII presents the calculated radionuclide activities released and peak concentrations within the containment building from a postulated maximum control rod ejection accident.

TABLE XXVIII

RADIOACTIVITY RELEASE TO CONTAINMENT BUILDING
FROM ROD EJECTION ACCIDENT
(Maximum Case)

Radionuclide	Radioactivity Released to Containment Building (Ci)	Peak Containment Airborne Radioactivity Concentration (μCi/cc)
Br-83	1.4(2)[a]	3.4(-3)
Br-84	2.5(2)	6.4(-3)
Br-85	2.7(2)	6.9(-3)
Kr-83m	2.7(2)	6.8(-3)
Kr-85m	6.7(2)	1.7(-2)
Kr-85	9.0(1)	2.3(-3)
Kr-87	1.3(3)	3.4(-2)
Kr-88	1.9(3)	4.8(-2)
Kr-89	2.0(3)	5.1(-2)
Xe-131m	1.4(1)	3.6(-4)
Xe-133m	9.2(1)	2.3(-3)
Xe-133	3.9(3)	9.7(-2)
Xe-135m	1.1(3)	2.7(-2)
Xe-135	3.6(3)	9.1(-2)
Xe-137	3.2(3)	8.1(-2)
Xe-138	3.3(3)	8.3(-2)
I-131	9.0(2)	2.3(-2)
I-132	1.3(3)	3.3(-2)
I-133	1.9(3)	4.9(-2)
I-134	2.3(3)	5.7(-2)
I-135	1.7(3)	4.4(-2)
Total	3.0(4)	7.6(-1)

[a] Example: 1.4(2) = 1.4×10^2.

5.2 Case 2 – Realistic Case

The assumptions differing from the maximum case are as follows:

(1) No additional fuel failure occurs as a result of the rod ejection

(2) The activity in the primary coolant is based on 0.12% of failed fuel[5].

Table XXIX presents a listing of the radionuclides released and their calculated activities resulting from steam dump from the secondary coolant system to the atmosphere for the assumed realistic rod ejection accident. Release of radioactivity was found to be independent of the type of steam generator plant (U-tube versus once-through) except for the halogens [see assumption 5.1(7) above]; however, the magnitude of halogen release was insignificant in both cases. The higher values (for once-through systems) are shown.

Table XXX presents the calculated radionuclide activities released and peak concentrations within the containment building resulting from a control rod ejection accident of assumed realistic magnitude.

TABLE XXIX

RADIOACTIVITY IN STEAM RELEASE TO ENVIRONMENT
FROM ROD EJECTION ACCIDENT
(Realistic Case)

Radionuclide	Radioactivity Released (Ci)
Kr-83m	8.8(-7)[a]
Kr-85m	5.6(-6)
Kr-85	8.6(-6)
Kr-87	2.2(-6)
Kr-88	9.2(-6)
Kr-89	4.7(-8)
Xe-131m	6.3(-6)
Xe-133m	1.2(-5)
Xe-133	1.0(-3)
Xe-135m	6.7(-7)
Xe-135	1.9(-5)
Xe-137	8.8(-8)
Xe-138	6.9(-7)
Br-83	6.2(-8)
Br-84	3.3(-8)
Br-85	3.8(-9)
I-130	3.5(-8)
I-131	3.5(-6)
I-132	1.3(-6)
I-133	4.9(-6)
I-134	6.0(-7)
I-135	2.4(-8)
Cr-51	2(-8)[b]
Mn-54	5(-9)
Fe-55	2(-8)
Fe-59	1(-8)
Co-58	2(-7)
Co-60	2(-8)
Rb-86	2(-9)
Rb-88	5(-6)
Sr-89	5(-9)
Sr-90	1(-10)
Sr-91	2(-9)
Y-90	2(-11)
Y-91m	5(-9)
Y-91	8(-10)
Y-93	5(-10)

TABLE XXIX (continued)

Radionuclide	Radioactivity Released (Ci)
Zr-95	8(-10)
Nb-95	5(-10)
Mo-99	1(-5)
Tc-99m	5(-6)
Ru-103	5(-10)
Ru-106	1(-10)
Rh-103m	5(-10)
Rh-106	1(-10)
Te-125m	3(-10)
Te-127m	3(-9)
Te-127	1(-8)
Te-129m	2(-8)
Te-129	2(-8)
Te-131m	1(-8)
Te-131	1(-8)
Te-132	3(-7)
Cs-134	5(-7)
Cs-136	3(-7)
Cs-137	4(-7)
Ba-137m	4(-7)
Ba-140	3(-9)
La-140	2(-9)
Ce-141	8(-10)
Ce-143	5(-10)
Ce-144	5(-10)
Pr-143	5(-10)
Pr-144	5(-10)
Np-239	2(-8)
Total	1(-3)

[a] Example: 8.8(-7) = 8.8×10^{-7}.

[b] Data accuracy to one significant digit.

TABLE XXX

RADIOACTIVITY RELEASED TO CONTAINMENT BUILDING
FROM ROD EJECTION ACCIDENT
(Realistic Case)

Radionuclide	Radioactivity Released to Containment Building (Ci)	Peak Containment Containment Airborne Radioactivity Concentration (μCi/cc)
Kr-83m	2.9	7.2(-5)[a]
Kr-85m	1.5(1)	3.8(-4)
Kr-85	2.0(1)	5.1(-4)
Kr-87	8.2	2.1(-4)
Kr-88	2.7(1)	6.9(-4)
Kr-89	6.8(-1)	1.7(-5)
Xe-131m	1.5(1)	3.8(-4)
Xe-133m	3.0(1)	7.5(-4)
Xe-133	2.4(3)	6.2(-2)
Xe-135m	1.8	4.5(-5)
Xe-135	4.8(1)	1.2(-3)
Xe-137	1.2	3.1(-5)
Xe-138	6.0	1.5(-4)
Br-83	3.3(-1)	8.2(-6)
Br-84	1.8(-1)	4.5(-6)
Br-85	2.0(-2)	5.1(-7)
I-130	1.4(-1)	3.6(-6)
I-131	1.8(1)	4.6(-4)
I-132	6.8	1.7(-4)
I-133	2.6(1)	6.5(-4)
I-134	3.2	8.1(-5)
I-135	1.3(1)	3.3(-4)
Cr-51	2.6(-1)	6.5(-6)
Mn-54	4.2(-2)	1.1(-6)
Fe-59	2.2(-1)	5.5(-6)
Co-58	1.4(-1)	3.4(-6)
Co-60	2.7(-1)	6.9(-6)
Rb-86	1.2(-2)	2.9(-7)
Rb-88	2.7(1)	6.9(-4)
Sr-89	4.8(-2)	1.2(-6)
Sr-90	1.4(-3)	3.4(-8)
Sr-91	8.8(-2)	2.2(-6)
Y-90	1.6(-4)	4.1(-9)
Y-91m	4.9(-2)	1.2(-6)
Y-91	8.7(-3)	2.2(-7)
Y-93	4.6(-3)	1.2(-7)

TABLE XXX (continued)

Radionuclide	Radioactivity Released to Containment Building (Ci)	Peak Containment Containment Airborne Radioactivity Concentration (μCi/cc)
Zr-95	8.2(-3)	2.1(-7)
Nb-95	6.8(-3)	1.7(-7)
Mo-99	1.1(1)	2.9(-4)
Tc-99m	6.5	1.6(-4)
Ru-103	6.1(-3)	1.5(-7)
Ru-106	1.4(-3)	3.4(-8)
Rh-103m	6.1(-3)	1.5(-7)
Rh-106	1.4(-3)	3.4(-8)
Te-125m	3.9(-3)	9.9(-8)
Te-127m	3.8(-2)	9.6(-7)
Te-127	1.2(-1)	2.9(-6)
Te-129m	1.9(-1)	4.8(-6)
Te-129	2.2(-1)	5.5(-6)
Te-131m	3.4(-1)	8.6(-6)
Te-131	1.5(-1)	3.8(-6)
Te-132	3.7	9.3(-5)
Cs-134	3.4	8.6(-5)
Cs-136	1.8	4.5(-5)
Cs-137	2.4	6.2(-5)
Ba-137m	2.4	6.2(-5)
Ba-140	3.0(-2)	7.5(-7)
La-140	2.0(-2)	5.1(-7)
Ce-141	9.5(-3)	2.4(-7)
Ce-143	5.4(-3)	1.4(-7)
Ce-144	4.5(-3)	1.1(-7)
Pr-143	6.8(-3)	1.7(-7)
Pr-144	4.5(-3)	1.1(-7)
Np-239	1.6(-1)	4.1(-6)
Total	2.7(3)	7.0(-2)
H-3	1.4(2)	3.4(-3)

[a] Example: 7.2(-5) = 7.2×10^{-5}.

X-C. ACCIDENT CLASS 8.0 – ACCIDENT INITIATION EVENTS CONSIDERED IN DESIGN BASIS EVALUATION IN THE SAFETY ANALYSIS REPORT

Accident 8.3(a) – Steamline Break Accidents

1. INTRODUCTION

The rupture of a portion of the secondary coolant system pressure boundary results in release of radioactivity in proportion to existing leakage from the primary coolant system to the secondary coolant system. The release of radioactivity for either a large (main steamline) or small (feedwater line) rupture depends on the primary-to-secondary coolant leakage rate, the percentage of defective fuel in the core, and the break size, which affects the duration of the blowdown. Since some locations at which ruptures could occur are outside the containment building, the radioactivity can be discharged directly to the environment.

2. POSTULATED CAUSES

The operation of the reactor with low concentrations of fission products in the secondary coolant system (requiring both defective fuel and steam generator tube leakage) is assumed.

The coincident failure of the secondary coolant system pressure boundary at a time when large quantities of fission products are available for release is considered to be unlikely. However, the failure of some pressure boundary components, such as valve packing, pump seals, and flange seals is probable during the life of the plant. Also, operation with defective fuel and coincident steam generator tube leakage is allowed by the plant technical specifications.

In order to bound the radiological release concentrations for secondary system pressure boundary failures, two cases are considered: one for a main steamline failure (large line) and the other for the equivalent of a feedwater line failure. The cases considered evaluate one failure outside the containment building (main steamline) and one inside the containment building (feedwater line).

A more probable release path for fission products in the secondary coolant system is through lifting of one or more of the relief valves. These valves will lift upon many anticipated operational transients (e.g., loss of load). Should a valve stick in the open position, a release of fluids and gases similar to the steamline rupture could occur from the open relief valve(s).

3. ACCIDENT DESCRIPTION

3.1.1 Major Secondary Coolant System Pipe Rupture (Rupture of a Main Steamline). The steam release arising from a rupture of main steam pipe would result in an initial increase in steam flow which subsequently decreases during the accident as the steam pressure falls.

Fast-acting isolation valves are provided in each steamline; these valves are designed to fully close within about 5 seconds of a large break in the steamline. For breaks downstream of the isolation valves, closure of the valves would terminate the blowdown. For any break, in any location, no more than one steam generator would blow down, even if one of the isolation valves failed to close.

Steam flow is usually measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles, which are of considerably smaller diameter than the main steam pipe, are located inside the containment building near the steam generators and also serve to limit the maximum steam flow for any break further downstream.

Two basic steamline failures could be considered in determining consequences of an accident:

(1) Complete severance of a pipe outside the containment building, downstream of the steam flow measuring nozzle

(2) Complete severance of a pipe inside the containment building at the outlet of the steam generator.

Although the effects of (2) may be more significant from the reactor safety aspect, the radiological environmental consequences of (1) are of more concern in this study, and therefore it is the only case considered.

The break assumed is the largest break which can occur anywhere outside the containment building either upstream or downstream of the isolation valves. The analysis assumes an uncontrolled steam release from one steam generator. The initiation of safety injection by high differential pressure between any steamline and the remaining steamlines, or by high steam flow signals in coincidence with either low-flow reactor coolant system temperature or low steamline pressure, will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steamlines by the high steam flow signals in coincidence with either low reactor coolant system temperature of low steamline pressure. The steamline isolation valves are designed to be fully closed in less than about 5 seconds after receipt of the closure signal[6].

3.1.2 Major Rupture of a Main Feedwater Pipe[6]. A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break in the containment building. Further a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedline check valve would affect the nuclear steam supply system only as a loss of feedwater and has no radiological consequences.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a primary coolant system cooldown (by excessive energy discharge through the break) or a primary coolant system heatup. Only the primary coolant system heatup effects are discussed for a feedline rupture since the cooldown effects are the same as for a main steam pipe rupture.

A feedline rupture would reduce the ability of the primary coolant system to remove heat generated by the core for the following reasons:

(1) Feedwater to the steam generators would be reduced. Since feedwater is subcooled, its loss could cause primary coolant temperatures to increase prior to reactor trip.

(2) Liquid in the steam generator could be discharged through the break, and would then not be available for decay heat removal after trip.

(3) The break could be large enough to prevent the addition of any main feedwater after trip.

Since protection systems for the postulated feedwater line rupture are adequate to remove decay heat, prevent overpressurizing the primary coolant system, and prevent uncovering the reactor core, the evaluation of this accident is for a release of fission products in the secondary system as a combined result of both defective fuel and steam generator tube leakage.

4. EFFLUENT RELEASE PATHWAY

4.1.1 Point of Maximum Concentration. The maximum concentration of airborne fission products would probably occur at the point of the release for a steamline or feedwater line break. It is not possible to identify a specific release pathway for pipe ruptures; however, the more probable release path would be the relief valve discharge line. For purposes of this study, calculations of radionuclide concentrations were made at the relief valve discharge location. For the assumed feedwater line rupture, the maximum concentration would occur within the containment building.

4.1.2 Principal Points of Environmental Release. For the assumed steamline break, the principal point of environmental release is assumed to be the relief valve discharge point; environmental release may also occur at the point of the break. For the assumed feedwater line break, environmental release can occur by several paths. One such path is release through the containment building ventilation system until the system is isolated. An alternate path is leakage from the isolated containment through penetrations such as airlocks and electrical penetrations.

5. ACCIDENT ANALYSIS ASSUMPTIONS AND CONDITIONS

5.1 Case 1 – Maximum Release Case (Large Steamline Break)

The assumptions and accident conditions for a large steamline break are in accordance with the general assumptions of NUREG-0099 (Regulatory Guide 4.2, Rev. 2) Appendix I[4], except where typical plant operating conditions indicate otherwise.

(1) The primary coolant activity is based on operation with 0.5% failed fuel[4] with the assumption that no further fuel failure results from the accident. The reactor is assumed to be operated at a core thermal power of 3800 MWt continuously for a period sufficient to establish equilibrium concentrations of the radioactive isotopes in the primary coolant (620 days).

(2) The secondary coolant system activity prior to the accident is based on a 100 lb/day primary-to-secondary leak rate.

(3) Prior to the accident, the iodine partition factor in a U-tube steam generator is assumed to be 0.1[3]. For a once-through steam generator, this factor is assumed to be 1.0[5]. Radionuclide concentrations in the secondary coolant prior to the accident were taken from Tables 2-2 and 2-3 of NUREG-0017, with appropriate corrections for iodine partition factor and for the assumed failed fuel rate of 0.5%.

(4) In that the worst case evaluated was that for a PWR with U-tube steam generators and phosphate water treatment, Tables XXXI and XXXII present releases from once-through steam generator plants and from U-tube steam generator plants with phosphate water treatment.

(5) The entire secondary coolant mass of one steam generator is assumed to be released to the atmosphere. For calculational purposes, the mass of the secondary coolant released was assumed to be 1.5×10^5 pounds.

TABLE XXXI

RADIOACTIVITY RELEASED TO ENVIRONMENT FROM LARGE STEAMLINE RUPTURE OUTSIDE CONTAINMENT BUILDING

	Radioactivity Released to Atmosphere (Ci)	
Radionuclide	U-tube (Phosphate)	Once-through
Kr-83m	1.6(-6)[a]	1.6(-6)
Kr-85m	8.8(-6)	8.8(-6)
Kr-85	1.2(-5)	1.2(-5)
Kr-87	4.5(-6)	4.5(-6)
Kr-88	1.6(-5)	1.6(-5)
Kr-89	4.0(-7)	4.0(-7)
Xe-131m	8.8(-6)	8.8(-6)
Xe-133m	1.8(-5)	1.8(-5)
Xe-133	1.4(-3)	1.4(-3)
Xe-135m	1.0(-6)	1.0(-6)
Xe-135	2.7(-5)	2.7(-5)
Xe-137	7.1(-7)	7.1(-7)
Xe-138	3.4(-6)	3.4(-6)
Br-83	4.2(-6)	6.5(-8)
Br-84	5.7(-7)	3.4(-8)
Br-85	5.7(-9)	4.0(-9)
I-130	7.1(-6)	3.7(-8)
I-131	3.1(-3)	3.7(-6)
I-132	3.1(-4)	1.3(-6)
I-133	1.8(-3)	5.1(-6)
I-134	1.6(-5)	6.2(-7)
I-135	4.0(-4)	2.5(-6)
Cr-51	2(-7)[b]	3(-7)
Mn-54	6(-8)	6(-8)
Fe-59	2(-7)	2(-7)
Co-58	1(-7)	1(-7)
Co-60	2(-6)	2(-6)
Rb-86	1(-8)	2(-8)
Rb-88	2(-7)	6(-5)
Sr-89	6(-8)	6(-8)
Sr-90	1(-9)	1(-9)
Sr-91	2(-8)	8(-8)
Y-90	6(-10)	2(-10)
Y-91m	8(-9)	6(-8)
Y-91	8(-9)	8(-9)
Y-93	1(-9)	6(-9)
Zr-95	8(-9)	8(-9)

TABLE XXXI (continued)

Radionuclide	Radioactivity Released to Atmosphere (Ci)	
	U-tube (Phosphate)	Once-through
Nb-95	8(-9)	6(-[illegible])
Mo-99	8(-6)	1(-4)
Tc-99m	8(-6)	6(-5)
Ru-103	6(-9)	6(-9)
Ru-106	1(-9)	1(-9)
Rh-103m	6(-9)	6(-9)
Rh-106	1(-9)	1(-9)
Te-125m	3(-9)	3(-9)
Te-127m	3(-8)	3(-8)
Te-127	6(-8)	1(-7)
Te-129m	2(-7)	2(-7)
Te-129	2(-7)	2(-7)
Te-131m	1(-7)	1(-7)
Te-131	1(-7)	1(-7)
Te-132	2(-6)	3(-6)
Cs-134	3(-6)	6(-6)
Cs-136	1(-6)	3(-6)
Cs-137	2(-6)	4(-6)
Ba-137m	2(-6)	2(-6)
Ba-140	3(-8)	3(-8)
La-140	2(-8)	2(-8)
Ce-141	8(-9)	8(-9)
Ce-143	3(-9)	6(-9)
Ce-144	6(-9)	6(-9)
Pr-143	6(-9)	6(-9)
Pr-144	8(-9)	6(-9)
Np-239	8(-8)	2(-7)
Total	7(-3)	2(-3)
H-3	3(-1)	3(-1)

[a] Example: $1.6(-6) = 1.6 \times 10^{-6}$.

[b] Data accuracy to one significant digit.

TABLE XXXII

RADIOACTIVITY RELEASED TO CONTAINMENT BUILDING
FROM A FEEDWATER LINE BREAK

	Radioactivity Released (Ci)		Peak Containment Airborne Radioactivity Concentration (μCi/cc)	
Radionuclide	U-tube (Phosphate)	Once-through	U-tube (Phosphate)	Once-through
Kr-83m	3.9(-7)[a]	3.9(-7)	9.9(-12)	9.9(-12)
Kr-85m	2.1(-6)	2.1(-6)	5.3(-11)	5.3(-11)
Kr-85	2.9(-6)	2.9(-6)	7.2(-11)	7.2(-11)
Kr-87	1.1(-6)	1.1(-6)	2.7(-11)	2.7(-11)
Kr-88	3.7(-6)	3.7(-6)	9.4(-11)	9.4(-11)
Kr-89	9.5(-8)	9.5(-8)	2.4(-12)	2.4(-12)
Xe-131m	2.1(-6)	2.1(-6)	5.3(-11)	5.3(-11)
Xe-133m	4.2(-6)	4.2(-6)	1.1(-10)	1.1(-10)
Xe-133	3.4(-4)	3.4(-4)	8.6(-9)	8.6(-9)
Xe-135m	2.4(-7)	2.4(-7)	6.2(-12)	6.2(-12)
Xe-135	6.6(-6)	6.6(-6)	1.7(-10)	1.7(-10)
Xe-137	1.7(-7)	1.7(-7)	4.3(-12)	4.3(-12)
Xe-138	8.1(-7)	8.1(-7)	2.1(-11)	2.1(-11)
Br-83	1.0(-6)	1.6(-8)	2.6(-11)	3.9(-13)
Br-84	1.4(-7)	8.1(-9)	3.4(-12)	2.1(-13)
Br-85	1.4(-9)	9.5(-10)	3.4(-14)	2.4(-14)
I-130	1.7(-6)	8.8(-9)	4.3(-11)	2.2(-13)
I-131	7.5(-4)	8.8(-7)	1.9(-8)	2.2(-11)
I-132	7.5(-5)	3.2(-7)	1.9(-8)	8.0(-12)
I-133	4.4(-4)	1.2(-6)	1.1(-8)	3.1(-11)
I-134	3.9(-6)	1.5(-7)	9.8(-11)	3.8(-12)
I-135	9.5(-5)	6.1(-7)	2.4(-9)	1.5(-11)
Cr-51	5(-8)[b]	6(-8)	1(-12)	2(-12)
Mn-54	1(-8)	1(-8)	3(-13)	3(-13)
Fe-59	3(-8)	5(-8)	9(-13)	1(-12)
Co-58	5(-7)	3(-8)	1(-11)	9(-13)
Co-60	6(-8)	5(-7)	2(-12)	1(-11)
Rb-86	3(-9)	5(-9)	7(-14)	1(-13)
Rb-88	5(-8)	1(-5)	1(-12)	3(-10)
Sr-89	1(-8)	1(-8)	3(-13)	3(-13)
Sr-90	3(-10)	3(-10)	9(-15)	9(-15)
Sr-91	4(-9)	2(-8)	1(-13)	5(-13)
Y-90	1(-10)	4(-11)	3(-15)	1(-15)
Y-91m	2(-9)	1(-8)	5(-14)	3(-13)
Y-91	2(-9)	2(-9)	5(-14)	5(-14)
Y-93	3(-10)	1(-9)	7(-15)	3(-14)

TABLE XXXII (continued)

Radionuclide	Radioactivity Released (Ci)		Peak Containment Airborne Radioactivity Concentration (μCi/cc)	
	U-tube (Phosphate)	Once-through	U-tube (Phosphate)	Once-through
Zr-95	2(-9)	2(-9)	5(-14)	5(-14)
Nb-95	2(-9)	1(-9)	5(-14)	3(-14)
Mo-99	2(-6)	3(-5)	5(-11)	7(-10)
Tc-99m	2(-6)	1(-5)	5(-11)	3(-10)
Ru-103	1(-9)	1(-9)	3(-14)	3(-14)
Ru-106	3(-10)	3(-10)	9(-15)	9(-15)
Rh-103m	1(-9)	1(-9)	9(-14)	3(-14)
Rh-106	3(-10)	3(-10)	9(-15)	9(-15)
Te-125m	6(-10)	7(-10)	2(-14)	2(-14)
Te-127m	6(-9)	7(-9)	2(-13)	2(-13)
Te-127	1(-8)	3(-8)	3(-13)	7(-13)
Te-129m	4(-8)	5(-8)	1(-12)	1(-12)
Te-129	4(-8)	5(-8)	1(-12)	1(-12)
Te-131m	3(-8)	3(-9)	9(-13)	9(-13)
Te-131	3(-8)	3(-8)	9(-13)	9(-13)
Te-132	5(-7)	7(-7)	1(-11)	2(-11)
Cs-134	8(-7)	1(-6)	2(-11)	3(-11)
Cs-136	3(-7)	7(-7)	9(-12)	2(-11)
Cs-137	5(-7)	1(-6)	1(-11)	3(-11)
Ba-137m	5(-7)	1(-6)	1(-11)	3(-11)
Ba-140	6(-9)	6(-9)	2(-13)	2(-13)
La-140	5(-9)	5(-9)	1(-13)	1(-13)
Ce-141	2(-9)	2(-9)	5(-14)	5(-14)
Ce-143	6(-10)	6(-10)	2(-14)	2(-14)
Ce-144	1(-9)	1(-9)	3(-14)	3(-14)
Pr-143	1(-9)	1(-9)	3(-14)	3(-14)
Pr-144	2(-9)	2(-9)	5(-14)	5(-14)
Np-239	2(-8)	2(-8)	5(-13)	5(-13)
Total	2(-3)	4(-4)	4(-8)	1(-8)
H-3	7(-2)	2(-6)	7(-2)	2(-6)

[a] Example: $3.9(-7) = 3.9 \times 10^{-7}$.

[b] Data accuracy to one significant digit.

Table XXXI presents the calculated radionuclide activities released to the atmosphere from a large steamline rupture outside the cc tainment building for both a U-tube steam generator plant (phosphate treatment) and a once-through steam generator plant.

5.2 Case 2 – Realistic Case (Small Break, Equivalent to a Feedwater Line Rupture)

All of the assumptions for the large break are used, with the following exceptions:

(1) Primary coolant activity is based on operation with 0.12% failed fuel[5]

(2) The activity is released to the containment building volume, assumed to be 2×10^6 ft^3

(3) Seventy percent mixing within the containment building is assumed[5].

Table XXXII presents the calculated radionuclide activities and peak containment building concentrations resulting from a postulated feedwater line break inside containment for a U-tube steam generator plant (phosphate treatment) and a once-through system.

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16. ABSTRACT (200 words or less)

Releases of gaseous radioactive materials were calculated for a series of postulated accidents for BWRs and PWRs. The purpose was the development of bases for selection of radiation measurement sensitivities for gaseous effluent monitoring instrumentation for accidents. The accidents considered were categorized according to the designations of NUREG-0099, Appendix I, (Regulatory Guide 4.2, Revision 2, July 1976).

Eight accident types in five BWR accident classes were considered, together with ten accident types in five PWR accident classes. Due to the budget limit it was not possible to include calculations for all postulated accidents; therefore, the study should not be considered as a definitive generic study for accident releases. Where postulated accident conditions projected releases into confined spaces, numerical values for peak concentrations are presented. Where releases were projected to be directly to the environment, total release quantities and individual contributions are presented. For accidents involving fuel damage, core radionuclide inventories were calculated using the ORIGEN (ORNL Isotope Generation and Depletion) code. Equilibrium inventories of radionuclides in primary and secondary cooling systems were obtained from NUREG-0016 (BWR) and NUREG-0017 (PWR).

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