15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 GASEOUS RADWASTE SYSTEM LEAK OR FAILURE

The following gaseous radwaste system components are examined under severe failure mode conditions for effects on the plant safety profile:

(1) Main condenser offgas treatment system failure

(2) Malfunction of main turbine gland sealing system

(3) Failure of Air Ejector Lines.

15.7.1.1 Ambient Charcoal Offgas Treatment System Failure

15.7.1.1.1 Identification of Causes and Frequency Classification

15.7.1.1.1.1 Identification of Causes

Those potential events which could cause a gross failure in the offgas treatment system are:

(1) A seismic occurrence - greater than design basis

The seismic event is considered to be the most probable and most severe which the system is designed to prevent or accommodate. The seismic failure is the only conceivable event which could cause significant system damage.

(2) A hydrogen explosion in housing unit

The equipment and piping are designed to contain any hydrogen-oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a failure mode which would cause significant system damage.

(3) A fire in the filter assemblies, and

The decay heat on the filters is easily handled inherently by the system and certainly by the available air flows.

(4) Failure of spatially related equipment.

The system is reasonably isolated from other systems or components which could cause any serious interaction or failure.

The only credible event which could result in the release of significant activity to the environment is an earthquake.

Even though the offgas system is designed to uniform building code seismic requirements, an event more severe than the design requirements is arbitrarily assumed to occur, resulting in the failure of the offgas system.

The design basis, description, and performance evaluation of the subject system is given in Section 11.3.

15.7.1.1.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.1.2 Sequence of Events and System Operation

15.7.1.1.2.1 Sequence of Events

The offgas treatment system is assumed to fail, resulting in releases from the offgas system charcoal adsorption beds, delay line, and the SJAE. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

Detection and reporting of a failure in the offgas treatment system will be performed in accordance with plant operating procedures.

The sequence of events following this failure is shown in Table 15.7-1.

15.7.1.1.2.2 Identification of Operator Actions

Upon verification of a failure, the operator may initiate a normal shutdown of the reactor to reduce the gaseous activity being discharged.

Gross failure of this system may require manual isolation of this system from the main condenser. This isolation results in high condenser pressure and a reactor scram. The operator will monitor the turbine generator auxiliaries and break vacuum as soon as possible. The operator must notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for reentry.

15.7.1.1.2.3 Systems Operation

In analyzing the postulated Offgas System failure, no credit is taken for the operation of plant and reactor protection systems, or of engineered safety features. Credit is taken for plant operating procedures and the functioning of normally operating plant instruments and controls and other systems only in assuming the following:

(1) Capability to detect the failure itself - indicated by an alarmed increase in radioactivity levels seen by Area Radiation Monitoring System, loss of flow in the Offgas System, and/or in an alarmed increase in activity at the vent release.

(2) Capability to isolate the system and shutdown the reactor.

(3) Operational indicator and annunciators in the main control room.

15.7.1.1.2.4 The Effect of Single Failures and Operator Errors

After the initial system gross failure, the inability of the operator to actuate a system isolation could affect the analysis.
However, the seismic event which is assumed to occur beyond the present plant design basis for non-safety equipment will undoubtedly cause the tripping of turbine or will lead to a load rejection. This will initiate a scram and negate a need for the operator to initiate a reactor shutdown via system isolation.

See Appendix 15A for a further discussion.

15.7.1.1.3 Core and System Performance

The postulated failure results in a system isolation necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Subsection 15.2.5.

15.7.1.1.4 Barrier Performance

The postulated failure is the rupture of the Offgas System pressure boundary. No credit is taken for performance of secondary barriers.

15.7.1.1.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions, for the purpose of determining adequacy of the plant design to limit the offsite doses to levels that are well within 10 CFR Part 50.67 guidelines. This analysis is referred to as the “design basis analysis.”

(2) The second is based on assumptions considered to provide a realistic yet conservative estimate of the radiological consequences. This analysis is referred to as the “realistic analysis.”

The radiological analyses assume that as a result of the offgas system failure, activity is released to the environment from the following sources:

- The total radioactive content of the offgas system charcoal adsorption beds.

- The total radioactive content of the offgas delay line.

- The release from the steam jet air ejector is assumed to occur from a break in the delay line just downstream of the SJAE. The SJAE is assumed to operate for a period of 1 hour after the accident. The release from the SJAE is assumed to be at ground level and a delay of 5 minutes is assumed to account for transit from the SJAE to the break in the delay line.

15.7.1.1.5.1 Design Basis Analysis

The gross failure of the offgas treatment system is assumed to release 100 percent of the noble gas inventory stored in the system with the continued release of offgas from the SJAE for a period of one hour. The radioactive content of the offgas during this period is conservatively based on an assumed 403,200 μCi/sec noble gas release rate at 30 minutes decay. The total activity released to the environment is given in Table 15.7-2. Specific parametric values used in this evaluation are presented in Table 15.7-5.

15.7.1.1.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The radiation source terms are based on a noble gas release rate from the SJAE equal to the design basis offgas release rate of 100,000 μCi/sec after 30 minutes delay. The system parameters used in determining the stored inventory and offgas releases are summarized in Table 15.7-5. The total activity released to the environment is given in Table 15.7-3.

15.7.1.1.5.3 Results

The calculated exposures offsite and at the control room for the design basis and realistic analyses of the Offgas Treatment System failure are presented in Table 15.7-4. A detailed description of the control room model is shown in Appendix 15B. The radiation dose consequences are well within the guideline values given in 10CFR50.67.

15.7.1.2 Malfunction of Main Turbine Gland Sealing System

15.7.1.2.1 Identification of Causes and Frequency Classification

15.7.1.2.1.1 Identification of Causes

Possible causes of the malfunction of the Turbine Gland Sealing System include the failure of the gland steam evaporator and its backup steam supply, failure of the gland steam condenser exhausters, and excessive pressure in the steam seal header.

15.7.1.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.2.2 Sequence of Events and System Operation

15.7.1.2.2.1 Sequence of Events

15.7.1.2.2.2 Identification of Operator Actions

It is assumed that the system fails near the condenser. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

The operator initiates a normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result in a turbine trip and reactor shutdown.

15.7.1.2.2.3 System Operation

Failure of the gland steam evaporator and its backup steam supply would result in the discharge of a small amount of contaminated steam from the HP and LP shaft seals to the gland steam condenser exhauster.

Failure of both of the gland steam condenser exhausters results in the escape of clean steam from the HP and LP shaft seals.

Excessive pressure in the steam seal header as a result of a malfunction of the gland steam evaporator or the backup steam supply valve is prevented by a relief valve so that there is no detrimental effect on the operation of the seals.

15.7.1.2.3 Core and System Performance

The failure of this power-conversion system does not directly affect the nuclear steam supply systems (NSSS). It will, of course, lead to decoupling of the NSSS with the power conversion system.

The tripping of the main turbine via main condenser signals will result in an anticipated operational transient examined earlier in Chapter 15.

This failure has no effect on the core or the NSSS safety performance.

15.7.1.2.4 Barrier Analysis

This release occurs outside the containment hence does not involve any barrier integrity aspects.

15.7.1.2.5 Radiological Consequences

Each of the assumed malfunctions results in negligible releases of activity. Therefore, the doses which result from these failures are inconsequential.

15.7.1.3 Failure of Air Ejector Lines

15.7.1.3.1 Identification of Causes and Frequency Classification

An evaluation of events that could cause a failure of the air ejector line indicates that a seismic event more serious than the system is designed to withstand is the only event that could rupture the lines. The lines are designed to withstand the effects of a hydrogen explosion.

This event is categorized as a limiting fault.

15.7.1.3.2 Sequence of Events and Systems Operation

The sequence of the events following this failure is shown in Table 15.7-18. It is assumed that the line leading from the steam jet air ejector to the offgas treatment system fails. This results in activity normally processed by the offgas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

Upon verification of a failure in the SJAE lines, the operator will initiate a normal shutdown of the reactor to reduce the gaseous activity being discharged. The operator will isolate the main condenser, which results in high condenser pressure and a reactor scram. The operator will notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for re-entry.

15.7.1.3.3 Core and System Performance

This auxiliary system does not directly affect the reactor core or the power cycle systems but is coupled only through operator alarms in the control room.

15.7.1.3.4 Barrier Analysis

This release occurs outside the containment; therefore, barrier integrity is not involved.

15.7.1.3.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions, similar to the offgas system failure, for the purpose of determining adequacy of the plant design to limit the offsite doses to levels that are well within 10 CFR Part 50.67 guidelines. This analysis is referred to as the “design basis analysis.”

(2) The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the “realistic analysis.”

15.7.1.3.5.1 Design Basis Analysis

For the design basis analysis, it is assumed that the reactor is operating at a steam flow of $1.69X10^{7}$ lb/hr, with the reactor coolant activity at design basis concentrations. The reactor steam concentrations of noble gases are based on an offgas release rate of 403,200 µCi/sec at 30 minutes decay. The iodine activity per pound of steam is assumed to be 8 percent of the iodine activity per pound of reactor coolant and an iodine partition factor of 100 exists between the condenser water and the offgas. The system parameters used in determining the stored inventory and offgas releases are summarized in Table 15.7-8.

15.7.1.3.5.2 Realistic Analysis

For the realistic analysis, it is assumed that the reactor is operating at a steam flow of 1.69×10^7 lb/hr, with the reactor coolant activity at design basis concentrations. The reactor steam concentrations of noble gases are based on an offgas release rate of 100,000 μCi/sec at 30 minutes decay. The iodine activity per pound of steam is assumed to be 8 percent of the iodine activity per pound of reactor coolant and an iodine partition factor of 140 exists between the condenser water and the offgas. The system parameters used in determining the stored inventory and offgas releases are summarized in Table 15.7-8.

15.7.1.3.5.3 Fission Product Release

The hypothetical failure of the air ejector lines is postulated to occur downstream of the hydrogen recombiner system. Automatic isolation of the system is not provided and it is conservatively assumed that the SJAE is isolated within 24 hours.

15.7.1.3.5.4 Fission Product Transport to the Environment

It is conservatively assumed that all the activity released from the SJAE line break is released to the environment, with no credit taken for plateout in the Turbine Building, prior to release to the environs.

These analyses assume no decay during transport and that the uncontrolled release period is 24 hours before the release is terminated. The total activity release rates for the design basis and realistic accidents are given in Table 15.7-6.

15.7.1.3.5.5 Results

The calculated exposures offsite and at the control room for the design basis and realistic analyses of the Steam Jet Air Ejector failure are presented in Table 15.7-7. A detailed description of the control room model is provided in Appendix 15B. The radiation dose consequences are well within the guideline values given in 10CFR50.67.

15.7.2 LIQUID RADWASTE SYSTEM FAILURE

15.7.2.1 Miscellaneous Small Releases Outside Containment

Releases that could occur from piping failures outside the containment include the feedwater system piping break (Subsection 15.6.6) and the main steam line break (Subsection 15.6.4) accidents. The analysis of these events provides doses that might occur for such a classification of piping failure events.

Releases to the environment that could occur from radwaste system component failures outside containment are addressed in Section 15.7.3 for radioactive gaseous releases to the atmosphere and in Sections 2.4.12 and 2.4.13 for radioactive liquid releases via the surface and groundwater pathways.

Other releases that could occur outside the containment include small spills and leaks of radioactive materials inside structures housing process equipment. Conservative values for leakage have been assumed and evaluated in Sections 11.2 and 11.3 under routine plant releases. The offsite dose that results from any small spill, which could occur outside the containment, will be negligible in comparison to the dose resulting from the postulated leakages.

Because the above references to other FSAR sections provide bounding uses for all conceivable small releases outside the containment, no further descriptions of core and system performance, barrier performance and radiological consequences are provided here.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID RADWASTE TANK FAILURE

15.7.3.1 Identification of Causes and Frequency Classification

15.7.3.1.1 Identification of Causes

An unspecified event causes complete release of the radioactive inventory in a liquid containing waste tank with the largest quantity of volatile radionuclides in the Radioactive Waste Management Systems. This component is the RWCU phase separator tank located in the radwaste enclosure. The airborne radioactivity released during this event is assumed to pass directly to the environment via the turbine building exhaust vent.

Postulated events that could cause release of the radioactive inventory of the RWCU phase separator are cracks in the vessels and operator error. The possibility of small cracks and consequently low-level release rates receives primary consideration in system and component design. The RWCU phase separator is designed to operate at atmospheric pressure and at a maximum temperature of 200°F so that the possibility of failure is considered small. A radioactive release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instructions. Should a liquid radioactive release occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid.

15.7.3.1.2 Frequency Classification

The complete rupture of the RWCU phase separator tank is considered a remote possibility. Although not analyzed for the requirements of Seismic Category I equipment, the Radioactive Waste Management System components are constructed in accordance with sound engineering principles. Therefore, simultaneous failure of all the tanks is not considered a credible event. Accordingly, this accident is expected to occur with the frequency of a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

The sequence of events expected to occur is as follows:

Sequence of Events

1. Event begins - failure occurs
2. Sump high level and or radiation alarms alert plant personnel
3. Operator actions begin

The rupture of the RWCU Phase Separator would leave little recourse to the operator. No method of recontaining the gaseous phase discharge is available, however isolation of the radwaste area would minimize personnel exposure. A high water level alarm in the radwaste building sump and radiation alarms in the turbine building exhaust vent and in the radwaste building area are available to alert the operator to the failure and followup actions would be taken to isolate the radwaste area ventilation system and proceed with cleanup operations. However, no credit for any operator action or for ventilation system isolation is assumed in evaluating the radiological consequences of this event.

15.7.3.3 Core and System Performance

The failure of the RWCU phase separator tank does not affect the Nuclear Steam Supply System (NSSS).

This failure has no applicable effect on the core or the NSSS safety performance.

15.7.3.4 Barrier Performance

This release occurs outside containment, therefore the event does not involve the primary containment barrier integrity.

15.7.3.5 Radiological Consequences

15.7.3.5.1 Design Basis Analysis

It is assumed that the RWCU phase separator tank contains the design basis inventory of radioactive material as presented in Table 15.7-9. The RWCU phase separator tank, which contains the largest amount of radioactive materials that could be released, is assumed to fail. The failure releases the entire contents of this tank to the radwaste enclosure. The radioactive materials in the RWCU phase separator tank are attached to powdered resin which is at ambient room temperature and are not expected to readily become airborne, if spilled. Consequently, the failure of the RWCU phase separator is not expected to result in any significant release of radioactive materials to the building atmosphere.

Nevertheless, a hypothetical event resulting in the release of radioactive iodine is evaluated. An iodine partition factor of 0.002 is assumed for the spilled liquid. This airborne iodine activity is vented through the radwaste ventilation system and exhausted instantaneously via the turbine building exhaust vent. No credit is assumed for iodine removal by the charcoal filters in the turbine building exhaust. Specific parametric values used in this evaluation are presented in Table 15.7-11. Table 15.7-9A lists the iodine activity assumed to be released to the environment. The offsite radiological doses for the RWCU phase separator rupture accident are given in Table 15.7-10.

15.7.3.5.2 Realistic Analysis

It is assumed that the inventory in the RWCU phase separator tank corresponds to an expected 0.05 Ci/sec offgas release rate at 30 minutes delay under normal operation as given in Table 15.7-9. Other parameters and assumptions are the same as those of the design basis analysis. Activities released to the environment and the offsite doses are presented in Tables 15.7-9A and 15.7-10, respectively.

15.7.4 FUEL AND EQUIPMENT HANDLING ACCIDENTS

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of the failure of the fuel assembly lifting mechanism resulting in the drop of a channeled fuel assembly, grapple, and mast onto other fuel bundles.

An equipment handling accident is assumed to occur as a consequence of the failure of the upper crane resulting in the drop of an object onto other fuel bundles. The total weight of the dropped object is 1100 lbs or less. Movement of objects in excess of 1000 lbs. are controlled by the Susquehanna Heavy Loads program.

A variety of events which qualify for the class of accidents termed "fuel and equipment handling accidents" have been investigated. The accidents which produce the most severe radiological consequences are the drop of a discharged channeled fuel assembly, grapple, and mast; or piece of equipment into the reactor core when the reactor vessel head is off or into the spent fuel pool.

Because the severity of the accident depends on a number of factors such as height of drop over impact site, depth of water over fuel (affects filtering of iodines), and recent irradiation/power history of fuel involved in impact (affects isotopic inventory or source term), a set of conservative assumptions is utilized to cover all possible scenarios. As such the fuel and equipment handling accident is analyzed to bound any credible event occurring over the core or over the spent fuel pool.

15.7.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

A typical sequence of events is as follows:

Event		**Approximate Elapsed Time**
(1)	A channeled fuel assembly is being handled over the core or spent fuel pool by the refueling equipment; or an object is being handled on the overhead crane over the core or spent fuel pool. The fuel assembly, grapple, and mast; or the object being handled on the overhead crane drops.	0
(2)	Fuel rods in the dropped fuel assembly and/or reactor core are damaged resulting in the release of gaseous fission products to the reactor coolant or spent fuel pool and eventually to the reactor building atmosphere.	0
(3)	The reactor building ventilation radiation monitoring system alarms to alert plant personnel, isolates the ventilation system, and starts operation of the Reactor Building Recirculation system and the SGTS.	1 min
(4)	Operator actions begin.	5 min

15.7.4.2.2 Identification of Operator Actions

(1) The operator will immediately initiate the evacuation of the refuel floor. If radiological conditions warrant, evacuation of the reactor building and the locking of the reactor building doors may also be initiated.

(2) The Refueling Floor Supervisor will instruct personnel to go immediately to the radiation protection personnel decontamination area.

(3) The Refueling Floor Supervisor will make the shift manager aware of the accident.

(4) The shift manager will immediately determine if the normal ventilation system has isolated, and the Reactor Building Recirculation system and the standby gas treatment system (SGTS) are in operation.

(5) The shift manager will initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the reactor building.

(6) The shift manager or his delegate will determine if the standby gas treatment system is performing as designed.

(7) The shift manager will post the appropriate radiological control signs at the entrance of the reactor building.

(8) Before entry to the reactor building is made, a careful study of conditions, radiation levels, etc., will be performed.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function although credit is taken only for the isolation of the normal ventilation system and the operation of the Reactor Building Recirculation System and the standby gas treatment system. Operation of other plant or reactor protection systems or ESF systems is not expected.

15.7.4.2.4 The Effects of Single Failures and Operator Errors

The automatic ventilation isolation system, which includes: a) the radiation monitoring detectors, b) isolation valves, and c) the Reactor Building Recirculation system and the SGTS are designed to single failure criteria and safety requirements.

Refer to Sections 7.6 and 9.4 and to Appendix 15A for further details.

15.7.4.3 Core and System Performance

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the mechanical consequences of the fuel and equipment handling accidents provide a conservative assessment of the number of fuel rods expected to fail.

For both the fuel handling and equipment handling accidents, a simple kinetic energy approach is used to determine fuel rod failures in the struck assemblies.

The fuel handling accident considers two impacts. First, an initial impact where the entire amount of kinetic energy is dispersed. After the initial impact, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally. The energy associated with the second impact is calculated by assuming a linear weight distribution over the length of the assembly and a point load at the top of the assembly representing the grapple and mast.

Half of the total kinetic energy available in each impact is assumed to be absorbed by the dropped fuel assembly, grapple, and mast. The dropped assembly is considered to impact at a small angle subjecting all the fuel rods in the dropped assembly to bending moments which result in the failure of all the fuel rods in the dropped assembly.

The other half of the total kinetic energy available in each impact is assumed to be absorbed by the non-fuel components of the struck fuel assemblies. This energy is then multiplied by the cladding weight fraction of the non-fuel components (weight of cladding to weight of total non-fuel components) of the struck fuel assemblies to determine the total amount of energy absorbed by the struck fuel rods.

The equipment handling accident considers one impact where the entire amount of available kinetic energy is absorbed by the struck fuel assemblies. This energy is then multiplied by the cladding weight fraction of the non-fuel components of the struck fuel assemblies to determine the total amount of energy absorbed by the struck fuel rods.

The total amount of energy absorbed by the struck fuel rods is divided by the fuel rod cladding failure energy to yield the expected number of failed fuel rods associated with each impact.

15.7.4.3.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are listed below:

(1) For the fuel handling accident a load of 1500 lbs. which conservatively represents the channeled fuel assembly, grapple, and mast is assumed to drop 32.95 ft (the maximum height that an irradiated fuel assembly can be carried) and impact other assemblies in the core or spent fuel pool.

For the equipment handling accident, a conservative load of 1100 lbs. is assumed to drop 150 ft (the maximum height that the overhead crane can carry an object) and impact onto fuel assemblies in the core or spent fuel pool.

(2) All of the fuel rods in the dropped fuel assembly (fuel handling accident) are conservatively assumed to fail as a result of the dropped fuel assembly being considered to impact at a small angle and being subjected to bending moments. Bending moments require significantly less energy (on the order of 1 ft-lb) to fail a fuel rod.

(3) It is assumed that no energy is absorbed by the uranium fuel material (UO_2) in the struck assemblies.

(4) It is assumed that no kinetic energy is dissipated in the water above the core or spent fuel pool.

(5) For the ATRIUM-11 fuel design the cladding weight fraction of the non-fuel components is approximately 0.4438.

(6) The energy required to produce cladding failure due to compression for an ATRIUM-11 fuel rod is approximately 160.9 ft-lbs. This is based upon a 1% plastic hoop strain in the rod.

15.7.4.3.3 Results

The results for fuel handling and equipment handling accidents involving both freshly discharged ATRIUM-10 and the ATRIUM-11 fuel are the most limiting of all the fuel types used in Susquehanna Units 1 and 2. The ATRIUM-10 and ATRIUM-11 case results also conservatively bound fuel designs in spent fuel pool, including the FANP 8x8, GE 8x8, and FANP 9x9-2 designs.

For each fuel type(s) specified below, the basis for why the ATRIUM-10 and ATRIUM-11 case is more limiting is provided:

GE 8x8 and FANP 8X8 fuel

Due to the extended decay time these fuel bundles have experienced since discharge from the reactor, the ATRIUM-10 and ATRIUM-11 source term will be larger.

FANP 9x9-2 fuel

Framatome has reported and documented that the ATRIUM-10 and ATRIUM-11 fuel and equipment handling accident is bounding over the 9x9-2. This is reasonable because the threshold to fail one ATRIUM-10 or one ATRIUM-11 fuel assembly is less than that to fail one 9x9-2 fuel assembly. Since the source term is about the same for an ATRIUM-10, ATRIUM-11, and 9x9-2 fuel assembly, the ATRIUM-10 and ATRIUM-11 fuel and equipment handling accidents would result in more assembly failures and a higher radiological release.

15.7.4.3.3.1 Energy Available

For the initial impact of the fuel handling accident, a load of 1500 lbs. representing the channeled fuel assembly, grapple, and mast is assumed to drop 32.95 ft and impact onto other fuel assemblies with a maximum kinetic energy of 49,425 ft-lbs. Following the initial impact, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally with a maximum kinetic energy of approximately 17,272 ft-lbs.

For the equipment handling accident, a load of 1100 lbs. is assumed to drop 150 ft and impact onto other fuel assemblies with a maximum kinetic energy of 165,000 ft-lbs.

15.7.4.3.3.2 Energy Loss Per Impact

Each impact is conservatively assumed to dissipate the total amount of kinetic energy available, with no credit taken for partial energy dissipation.

15.7.4.3.3.3 Fuel Rod Failures

For the purpose of determining the radiological consequences due to the postulated fuel handling accident case, the estimated number of failed fuel rods is 191 rods, including 79 failed fuel rods from ATRIUM-10 fuel assemblies and 112 failed fuel rods from an ATRIUM-11 fuel assembly. For the equipment handling accident case, the number of failed rods is estimated as 455 rods for the ATRIUM 11 fuel assemblies.

15.7.4.3.3.3.1 First Impact Failures

For the Fuel Handling Accident, the fuel rod failure calculations assume that the dropped fuel assembly is an ATRIUM-11 assembly and the struck fuel assemblies are ATRIUM-10 assemblies. As noted in Section 15.7.4.3.3 this provides the most limiting fuel handling and equipment handling accident results.

For the initial impact of the fuel handling accident, a total kinetic energy of 49,425 ft-lbs. is dissipated.

Half of the energy is assumed to be absorbed by the dropped fuel assembly, grapple, and mast. However, all of the fuel rods in the dropped assembly are conservatively assumed to fail as a result of impacting at a small angle and being subjected to bending moments.

The other half of the energy is assumed to be absorbed by the non-fuel components of the struck fuel assemblies. Framatome has reported and documented that this impact yields approximately 59 failed rods in the struck fuel assemblies. Thus, the first impact of the fuel handling accident yields the following fuel failures:

Dropped Assembly	112	rods (all rods assumed to fail)
Struck Assemblies	59	rods (1st impact)
	171	rods

For the impact of the equipment handling accident, a total kinetic energy of 165,000 ft-lbs is dissipated. The total kinetic energy is assumed to be absorbed by the non-fuel components of the struck assemblies. The ATRIUM-11 assembly non-fuel components cladding weight fraction is 0.4438. Therefore the total amount of energy absorbed by the struck fuel rods is approximately 73,277 ft-lbs. Dividing this by the cladding failure threshold of 160.9 ft-lbs yields approximately 455 failed rods in the struck fuel assemblies. Thus, the impact of the equipment handling accident yields the following fuel failures:

Struck Assemblies	455	rods

15.7.4.3.3.3.2 Second Impact Failures

Following the initial impact in the fuel handling accident, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally with a maximum kinetic energy of approximately 17,272 ft-lbs.

Half of that energy is assumed to be absorbed by the non-fuel components of the struck assemblies. Framatome has reported and documented that this impact yields approximately 20 failed rods in the struck fuel assemblies. Thus, the second impact of the fuel handling accident yields the following fuel failures:

Struck Assemblies	20	rods (2nd Impact for the limiting case)

15.7.4.3.3.3.3 Total Failures

The total number of failed rods resulting from the fuel handling accident is as follows:

First impact	171	rods
Second impact	20	rods
	191	total failed rods (2.01 assemblies)*

The total number of failed rods resulting from the equipment handling accident is as follows:

First impact	455	total failed rods (4.50 assemblies)*

*It is important to consider the total number of assemblies that fail because fuel designs have different numbers of rods. Consideration on a failed assembly basis provides a better measure of the relative severity of fuel and equipment handling accidents involving different mechanical designs because core average source terms do not account for assembly mechanical differences, and on an assembly basis are approximately the same.

15.7.4.4 Barrier Performance

The reactor coolant pressure boundary and primary containment are assumed to be open. The transport of fission products from the reactor building is discussed in Subsections 15.7.4.5.2.1 and 15.7.4.5.2.2 below.

15.7.4.5 Radiological Consequences

Two separate radiological analyses are provided for each refueling accident scenario:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR Part 50.67 guidelines. This analysis is referred to as the "Design Basis Analysis."

(2) The second analysis is based on assumptions considered to provide a realistic but still conservative estimate of radiological consequences. This analysis is referred to as the "Realistic Analysis."

For the Design Basis and Realistic analyses, the fission product inventory in the fuel rods assumed to be damaged is based on an average assembly burnup of 39,000 MWd/MTU for ATRIUM-10 fuel assemblies and 41,000 MWd/MTU for ATRIUM-11 fuel assemblies resulting from continuous operation at 4032 MW(t).

A 24-hour period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 hours following initiation of reactor shutdown. Figure 15.7-1 indicates the leakage flow path for these accidents.

15.7.4.5.1 Design Basis Analysis

The design basis analysis is based on NRC Regulatory Guide 1.183. The RADTRAD computer code is used to evaluate the radiological consequences (Reference 15.7-2). Specific values of parameters used in the evaluation for the fuel handling and equipment handling accidents are presented in Table 15.7-17.

Regulatory Guide 1.183 provides guidance on the use of pool decontamination factors for iodine for water depths of 23 feet or greater. For Susquehanna, the minimum water depth occurs over the spent fuel pool and is approximately 22 feet (for analysis purposes 21 feet is assumed, which is conservative). Regulatory Guide 1.183 states that if the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method. Regulatory Guide 1.183 indicates an acceptable method is provided in Staff Technical Paper, Evaluation of Fission Product Release and Transport, G. Burley (Reference 15.7-5). The overall decontamination factor for a pool depth of 21 feet determined using this methodology is a factor of 138.

15.7.4.5.1.1 Fission Product Release from Fuel

The fission product inventory of a core average rod (for radiological source term purposes, a core average rod for the ATRIUM-10 / ATRIUM-11 assembly is conservatively based upon 87.8 / 101.2 equivalent full-length fuel rods per assembly) is adjusted by a peaking factor of 1.6 to establish the inventory of each damaged rod. The analysis is bounding for the mixture of ATRIUM-10 / ATRIUM-11. The activity in the fuel rod gap available for release from the damaged rods is defined in accordance with Regulatory Guide 1.183 as eight percent of the I-131 inventory, 10 percent of the Kr-85 inventory, five percent of the noble gases and halogens and twelve percent of the alkali metals. These release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. A pool decontamination factor of 138 for iodine and 0 for noble gases is assumed. The activity airborne in the secondary containment is presented in Table 15.7-12 for both the fuel handling and equipment handling accident scenarios.

15.7.4.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of mixing in the fuel pool, migration from the pool to the secondary containment atmosphere and release to the environment through the SGTS (Standby Gas Treatment System).

After filtration by the SGTS (99% removal efficiency for iodine, 0% for noble gases, no filtration is assumed during the 10 minute drawdown) the airborne activity is assumed to be released to the environment over a 2 hour period. The dose over a 2-hour period from the start of the release is calculated at the exclusion area boundary. In addition, the dose over the 30-day period is calculated for the low population zone and the control room. No credit is taken for isotopic decay during the release.

The release of activity to the environment is presented in Table 15.7-13 for both the equipment and fuel handling accident scenarios.

15.7.4.5.1.3 Results

OFFSITE DOSES

The calculated radiological doses at the exclusion area boundary and low population zone for the design basis analyses are presented in Table 15.7-16. All doses are well within the 10CFR50.67 dose limits and the Regulatory Guide 1.183 acceptance criteria.

CONTROL ROOM DOSES

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.7-17. The radiological exposure to the control room personnel for the design basis case is given in Table 15.7-16. The calculated dose meets the 10CFR50.67 control room dose acception criterion.

15.7.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The RADTRAD computer code (Reference 15.7-2) is used to evaluate the radiological consequences of the realistic analyses. Specific values of parameters used in the evaluation for the equipment handling and fuel handling accidents are presented in Table 15.7-17.

15.7.4.5.2.1 Fission Product Release from Fuel

Fission product release estimates for the refueling accidents are based on the following assumptions:

(1) The reactor fuel has an average irradiation 41 GWD/MTU of up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of biologically significant isotopes. The 24-hr decay period allows time to shut down the reactor, depressurize the nuclear system, remove the reactor vessel head, and remove the reactor vessel upper internals. It is not expected that these operations could be accomplished in less than 24 hr and probably will require at least 48 hrs.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments (Reference 15.7-3).

(3) Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

(4) It is conservatively assumed that the same number of fuel rods fail for the realistic equipment handling and fuel handling accidents as used in the design basis analysis. This is considered to be conservative because it is expected that many fewer rods would be damaged for these accident scenarios.

15.7.4.5.2.2 Fission Product Transport to the Environment

The following assumptions and conditions are used in calculating the release of activity to the environment for the equipment and fuel handling accidents.

(1) All of the noble gases released to the fuel pool become airborne in the secondary containment (reactor building).

(2) A pool decontamination factor of 138 is used for iodine activity released from the fuel.

(3) All of the activity is released from the secondary containment to the environment through the SGTS in two (2) hours (99% removal efficiency for iodine assumed after a 10 minute drawdown, no filtration is assumed during drawdown).

Based on these assumptions, the activity airborne in the reactor building for each refueling accident scenario is shown in Table 15.7-14.

The release rate of activity under normal ventilation conditions is sufficient to cause a trip of the Secondary Containment Discharge Plenum radiation monitors which results in secondary containment isolation and SGTS startup.

The cumulative release to the environment is presented in Table 15.7-15.

15.7.4.5.2.3 Results

OFFSITE DOSES

The calculated exposures for the realistic analyses are presented in Table 15.7-16 for both refueling accidents and demonstrate the margin of conservatism in the design basis analysis.

CONTROL ROOM DOSES

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.7-17. The radiological exposure to the control room personnel for the realistic basis case is given in Table 15.7-16.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

The spent fuel cask will be equipped with redundant sets of lifting lugs and yokes compatible with the reactor building crane main hook, thus preventing a cask drop due to a single failure. Therefore, an analysis of the spent fuel cask drop is not required. The On-Site Transfer Cask (described in Section 11.7.6.1) and yoke used to transfer spent fuel to the Independent Spent Fuel Storage Installation (ISFSI) is single failure proof and compatible with the Unit 1 Reactor Building Crane main hook, thus preventing an On-Site Transfer Cask drop due to single failure. Therefore, an analysis of the On-Site Transfer Cask drop is not required. Refer to Subsection 9.1.5 for a description of the reactor building crane and the interlocks which prevent moving the spent fuel cask over the fuel pool.

15.7.6 REFERENCES

15.7-1 ORNL-4628, ORIGEN - The ORNL Isotope Generation and Depletion Code, March 1974.

15.7-2 NUREG/CR-6604 and Supplements, RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation, June 1999.

15.7-3 N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," APED 5756, March 1969.

15.7-4 "General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation, and Design Application," NEDO-10958 and NEDE-10958, (November 1973).

15.7-5 Staff Technical Paper, Evaluation of Fission Product Release and Transport, G. Burley, 1971 (NRC Accession Number 8402080322)

15.7-6 PL-NF-97-003, Rev. 1, “Susquehanna SES Unit 2 Cycle 9 Reload Summary Report,” September 1997.

15.7-7 PL-NF-96-005, Rev. 2, “Susquehanna SES Unit 1 Cycle 10 Reload Summary Report,” July 1997.

TABLE 15.7-1

SEQUENCE OF EVENTS FOR MAIN CONDENSER OFFGAS TREATMENT SYSTEM FAILURE

Elapsed Time	Events
0 seconds	Event begins with system failure and the release of radioactive gases to the building.
0 to 1 hour	Event detection and termination of release: • Event detection is based on an alarmed increase in activity release via the building vent monitor. • Area radiation monitor alarms. • Loss of flow indication in the offgas system. Operator actions include: • Verification and assessment of the accident. • Notifications of plant personnel for area evacuation and radiation protection/area survey. • Initiate appropriate system isolations. • Manual scram actuation. • Assurance of reactor shutdown cooling.

TABLE 15.7-2

ACTIVITY INVENTORY STORED IN OFFGAS TREATMENT SYSTEM[1]
ACTIVITY RELEASED TO THE ENVIRONS (curies)
(DESIGN BASIS ANALYSIS)

Isotope	Activity
I-131	1.31E-01
I-132	1.45E+00
I-133	9.24E-01
I-134	3.42E+00
I-135	1.41E+00
Kr-83m	1.84E+02
Kr-85m	6.71E+02
Kr-85	1.51E+02
Kr-87	8.40E+02
Kr-88	1.51E+03
Kr-89	6.79E+02
Xe-131m	6.12E+01
Xe-133m	3.24E+02
Xe-133	2.19E+04
Xe-135m	4.50E+02
Xe-135	4.55E+03
Xe-137	9.73E+02
Xe-138	1.48E+03

1. SJAE, delay pipe, and offgas system delay beds.

TABLE 15.7-3

ACTIVITY INVENTORY STORED IN OFFGAS TREATMENT SYSTEM[1]
ACTIVITY RELEASED TO THE ENVIRONS (curies)
(REALISTIC ANALYSIS)

Isotope	Activity
I-131	2.33E-02
I-132	2.58E-01
I-133	1.64E-01
I-134	6.06E-01
I-135	2.50E-01
Kr-83m	4.01E+01
Kr-85m	1.53E+02
Kr-85	4.67E+00
Kr-87	1.73E+02
Kr-88	3.39E+02
Kr-89	1.61E+02
Xe-131m	1.99E+01
Xe-133m	7.75E+01
Xe-133	5.36E+03
Xe-135m	9.16E+01
Xe-135	1.12E+03
Xe-137	2.27E+02
Xe-138	3.03E+02

1. SJAE, delay pipe, and offgas system delay beds.

TABLE 15.7-4

MAIN CONDENSER OFFGAS TREATMENT SYSTEM FAILURE RADIOLOGICAL EFFECTS

Accident Type	EAB (2 hr) (REM TEDE)	LPZ (duration) (REM TEDE)	CRHE (duration) (REM TEDE)
Realistic Analysis	4.38E-02	1.62E-03	1.68E-02
Design Basis Analysis	1.19E+00	7.02E-02	7.19E-02

TABLE 15.7-5

OFFGAS TREATMENT SYSTEM FAILURE
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

		Design Basis Analysis	Realistic Analysis
I.	**Data and Assumptions Used to Estimate Radioactive Sources from Postulated Accidents**		
A.	Reactor power (MWt)	4032	4032
B.	Fuel damage	None	None
C.	Reactor coolant activity before the accident		
	1. Iodine	-	Table 11.1-2
	2. Noble gas	Table 11.1-1	Table 11.1-1
D.	Reactor steam offgas release rate at 30 minutes decay (μCi/sec)	403,200	100,000
E.	Iodine carry over fraction reactor water to steam (percent)	8	8
II.	**Data and Assumptions Used to Estimate Activity Released to the Environment**		
A.	Total mass of charcoal in absorbers (lbs)	148,000	148,000
B.	Offgas system delay bed release to environs duration (hrs)	2	2
C.	Offgas system delay line release to environs duration (hrs)	2	2
D.	SJAE release to environs duration (hrs)	1	1
E.	Condenser air in-leakage / Common offgas recombiner low flow	6	21.76
	purge air (scfm)	1.46E+07	1.46E+07
F.	Reactor steam flow (lbm/hr)	65 – Kr	36 – Kr
G.	Dynamic absorption coefficients for the charcoal beds (cm^3/gm)	1000 – Xe	516 – Xe
III.	**Data and Assumptions Used to Evaluate Control Room Doses**		
A.	Control structure habitability envelope free volume(ft^3)	518,000	549/4827
B.	Control room free volume (ft^3)	110,000	110,000
C.	Control structure filtered air intake flow (cfm)	5229 – 6391	5229 - 6391
D.	Control structure unfiltered outside air infiltration rate –	10	10
	ingress/egress (cfm)	500	500
E.	Control structure unidentified unfiltered outside air infiltration rate	99	99
	(cfm)		
F.	Control structure filter efficiency for iodine (percent)		

TABLE 15.7-5

**OFFGAS TREATMENT SYSTEM FAILURE
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS**

IV.	**Dispersion Data**		
A.	EAB and LPZ distance (meters)	549/4829	549/4829
B.	EAB/XQ	Table 2.3-92 (0.5percentile)	Table 2.3-92 (0.5percentile)
C.	LPZ/QZ	Table 2.3-105 (0.5percentile)	Table 2.3-105 (0.5percentile)
D.	CRHE X/Q	Appendix 15B	Appendix 15B
V.	**Dose Date**		
A.	Method of dose calculations	Appendix 15B	Appendix 15B
B.	Dose conversion assumptions	Appendix 15B	Appendix 15B
C.	Doses	Table 15.7-4	Table 15.7-4

1. Times earlier than 30 days before the postulated accident is 100,000 μCi/sec at 30 minutes decay.

TABLE 15.7-6

FAILURE OF AIR EJECTOR LINES
ACTIVITY RELEASED TO THE ENVIRONMENT (curies/sec)

Isotope	Realistic Analysis	Design Basis Analysis
I-131	5.17E-06	2.92E-05
I-132	5.93E-05	3.35E-04
I-133	3.65E-05	2.06E-04
I-134	1.48E-04	8.33E-04
I-135	5.63E-05	3.18E-04
Kr-83m	3.40E-03	1.37E-02
Kr-85m	6.10E-03	2.46E-02
Kr-85	2.00E-05	8.06E-05
Kr-87	2.00E-02	8.06E-02
Kr-88	2.00E-02	8.06E-02
Kr-89	1.30E-01	5.24E-01
Xe-131m	1.50E-05	6.05E-05
Xe-133m	2.90E-04	1.17E-03
Xe-133	8.20E-03	3.31E-02
Xe-135m	2.60E-02	1.05E-01
Xe-135	2.20E-02	8.87E-02
Xe-137	1.50E-01	6.05E-01
Xe-138	8.90E-02	3.59E-01

TABLE 15.7-7

FAILURE OF STEAM JET AIR EJECTOR LINES
RADIOLOGICAL CONSEQUENCES

Accident Type	EAB (2 hr) (REM TEDE)	LPZ (duration) (REM TEDE)	CRHE (duration) (REM TEDE)
Realistic Analysis	2.86E-02	1.09E-02	6.79E-02
Design Basis Analysis	2.06E+00	1.18E+00	7.23E-01

TABLE 15.7-8

FAILURE OF AIR EJECTOR LINES
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Analysis	Realistic Analysis
I. Data and Assumptions Used to Estimate Radioactive Sources from Postulated Accidents		
A. Reactor power (MWt)	4302	4302
B. Fuel damage	None	None
C. Reactor coolant activity before the accident		
1. Iodine	Table 11.1-2	Table 11.1-2
2. Noble gas	Table 11.1-1	Table 11.1-1
D. Reactor steam offgas release rate at 30 minutes decay (µCi/sec)	403,200	100,000
E. Iodine carry over fraction reactor water to steam (percent)	8	8
II. Data and Assumptions Used to Estimate Activity Released to the Environment		
A. Condenser partition coefficients for iodines:	100	140
B. Time for system isolation (hrs)	24	24
C. Reactor steam flow (lbm/hr)	169E+07	169E+07
III. Data and Assumptions Used to Evaluate Control Room Doses		
A. Control structure habitability envelope free volume(ft^3)	518,000	518,000
B. Control room free volume(ft^3)	110,000	110,000
C. Control structure filtered air intake flow (cfm)	5229 - 6391	5229 - 6391
D. Control structure unfiltered outside air infiltration rate –	10	10
ingress/egress (cfm)	500	500
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	99	99
F. Control structure filter efficiency for iodine (percent)		
IV. Dispersion Data		
A. EAB and LPZ distance (meters)	549/4827	549/4827
B. EAB X/Q	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (0.5 percentile)
C. LPZ/ X/Q	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (0.5 percentile)
D. CRHE X/Q	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of dose calculations	Appendix 15B	Appendix 15B
B. Dose conversion assumptions	Appendix 15B	Appendix 15B
C. Doses	Table 15.7-7	Table 15.7-7

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TABLE 15.7-9

RWCU PHASE SEPARATOR TANK FAILURE – INITIAL ACTIVITY

Design Basis Analysis					
Isotope	**Activity (Ci)**	**Isotope**	**Activity (Ci)**	**Isotope**	**Activity (Ci)**
Ba-139	4.60E+01	I-134	4.32E+01	Sr-89	7.88E+02
Ba-140	5.83E+02	I-135	1.81E+02	Sr-90	3.09E+02
Ce-141	1.74E+01	La-140	5.84E+02	Sr-91	1.38E+02
Ce-143	2.44E-01	La-141	1.09E+01	Sr-92	6.25E+01
Ce-144	3.08E+01	La-142	6.41E+00	Tc-99m	6.25E+02
Co-58	1.75E+03	Mo-99	3.07E+02	Te-129	4.27E+00
Co-60	6.36E+02	Nb-95	2.00E+01	Te-129m	6.80E+00
Cs-134	1.83E+02	Nd-147	7.79E-01	Te-132	8.12E+02
Cs-136	7.35E+00	Np-239	2.86E+03	Y-91	1.37E+02
Cs-137	3.23E+02	Pr-143	2.86E+00	Y-92	6.29E+01
I-131	5.32E+02	Pu-239	8.67E-02	Zr-95	1.28E+01
I-132	8.68E+02	Ru-103	3.78E+00	Zr-97	1.14E-01
I-133	3.91E+02	Ru-106	2.51E+00	-	-

Realistic Analysis					
Isotope	**Activity (Ci)**	**Isotope**	**Activity (Ci)**	**Isotope**	**Activity (Ci)**
Ba-139	2.02E+00	I-135	2.41E+01	Sr-90	5.68E+00
Ba-140	1.62E+01	La-140	1.62E+01	Sr-91	5.17E+00
Ce-141	3.05E+00	La-142	1.15E+00	Sr-92	3.85E+00
Ce-143	1.32E-01	Mo-99	1.75E+01	Tc-99m	3.17E+01
Ce-144	1.60E+00	Nb-95	2.40E+00	Te-129	1.27E+00
Co-58	4.31E+01	Nd-147	1.04E-01	Te-129m	4.20E+00
Co-60	3.07E+02	Np-239	5.25E+01	Te-131m	8.01E-01
Cs-134	2.07E+01	Pr-143	1.85E+00	Te-132	1.04E-01
Cs-136	3.34E+00	Pu-239	1.54E-03	Y-91	1.22E+01
Cs-137	1.62E+01	Rh-105	1.25E+00	Y-92	6.90E+00
I-131	4.60E+01	Ru-103	2.46E+00	Y-93	5.51E+00
I-132	1.20E+01	Ru-105	1.24E+00	Zr-95	1.57E+00
I-133	6.94E+01	Ru-106	1.75E+00	-	-
I-134	9.06E+00	Sr-89	1.56E+01	-	-

TABLE 15.7-9A

RWCU PHASE SEPARATOR TANK FAILURE ACTIVITY RELEASE TO ENVIRONMENT

Realistic Source		Design Basis Source	
Isotope	**Curies**	**Isotope**	**Curies**
Ba-139	4.04E-03	Ba-139	9.20E-02
Ba-140	3.24E-02	Ba-140	1.17E+00
Ce-141	6.10E-03	Ce-141	3.48E-02
Ce-143	2.64E-04	Ce-143	4.88E-04
Ce-144	3.20E-03	Ce-144	6.16E-02
Co-58	8.62E-02	Co-58	3.50E+00
Co-60	6.14E-01	Co-60	1.27E+00
Cs-134	4.14E-02	Cs-134	3.66E-01
Cs-136	6.68E-03	Cs-136	1.47E-02
Cs-137	3.24E-02	Cs-137	6.46E-01
I-131	9.20E-02	I-131	1.06E+00
I-132	2.40E-02	I-132	1.74E+00
I-133	1.39E-01	I-133	7.82E-01
I-134	1.81E-02	I-134	8.64E-02
I-135	4.82E-02	I-135	3.62E-01
La-140	3.24E-02	La-140	1.17E+00
La-142	2.30E-03	La-141	2.18E-02
Mo-99	3.50E-02	La-142	1.28E-02
Nb-95	4.80E-03	Mo-99	6.14E-01
Nd-147	2.08E-04	Nb-95	4.00E-02
Np-239	1.05E-01	Nd-147	1.56E-03
Pr-143	3.70E-03	Np-239	5.72E+00
Pu-239	3.08E-06	Pr-143	5.72E-03
Rh-105	2.50E-03	Pu-239	1.73E-04
Ru-103	4.92E-03	Ru-103	7.56E-03
Ru-105	2.48E-03	Ru-106	5.02E-03
Ru-106	3.50E-03	Sr-89	1.58E+00
Sr-89	3.12E-02	Sr-90	6.18E-01
Sr-90	1.14E-02	Sr-91	2.76E-01
Sr-91	1.03E-02	Sr-92	1.25E-01
Sr-92	7.70E-03	Tc-99m	1.25E+00
Tc-99m	6.34E-02	Te-129	8.54E-03
Te-129	2.54E-03	Te-129m	1.36E-02
Te-129m	8.40E-03	Te-132	1.62E+00
Te-131m	1.60E-03	Y-91	2.74E-01
Te-132	2.08E-04	Y-92	1.26E-01
Y-91	2.44E-02	Zr-95	2.56E-02
Y-92	1.38E-02	Zr-97	2.28E-04
Y-93	1.10E-02		
Zr-95	3.14E-03		

TABLE 15.7-10

RWCU PHASE SEPARATOR TANK FAILURE RADIOLOGICAL EFFECTS

Accident Type	EAB (2 hr) (REM TEDE)	LPZ (duration) (REM TEDE)	CRHE (duration) (REM TEDE)
Realistic Analysis	7.53E-03	2.78E-04	6.28E-02
Design Basis Analysis	4.09E-01	2.42E-02	5.34E-01

TABLE 15.7-11

RWCU PHASE SEPARATOR TANK FAILURE

	Design Basis Assumptions	Realistic Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents		
A. Reactor power (MWt)	4032	4032
B. Fuel damaged	None	None
C. Reactor coolant activity before the accident		
Realistic	N/A	Table 15.7-9A
Conservative case	Table 15.7-9A	N/A
II. Data and Assumption Used to Estimate Activity Released		
A. Holdup in Radwaste Building	No	No
B. Particulate/Iodine Partition Coefficient	0.002	0.002
III. Data and Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume(ft^3)	518,000	518,000
B. Control room free volume(ft^3)	110,000	110,000
C. Control structure filtered air intake flow(cfm)	5229 - 6391	5229 - 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	500	500
F. Control structure filter efficiency for iodine (percent)	99	99
IV. Disposition Data		
A. Boundary for SB/LPZ distance (meters)	549/4827	549/4827
B. X/Q's for Site Boundary	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (50 percentile)
C. X/Q's for LPZ	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (50 percentile)
D. X/Q's for CRHE	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of dose calculation	Appendix 15B	Appendix 15B
B. Dose conversion assumptions	Appendix 15B	Appendix 15B
C. Doses	Table 15.7-10	Table 15.7-10

TABLE 15.7-12

REFUELING ACCIDENTS
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Design Basis Accident)

Isotope	Reactor Building Airborne Activity (curies)	
	Equipment Handling Accident (455 Failed Rods)	Fuel Handling Accident (191 Failed Rods)
I-131	5.51E+02	2.45E+02
I-132	4.37E+02	1.95E+02
I-133	3.47E+02	1.55E+02
I-134	1.90E-05	8.58E-06
I-135	5.74E+01	2.56E+01
Kr-83m	2.60E+01	1.13E+01
Kr-85m	3.25E+02	1.42E+02
Kr-85	1.57E+03	6.66E+02
Kr-87	5.27E-02	2.36E-02
Kr-88	9.66E+01	4.37E+01
Xe-131m	5.51E+02	2.72E+02
Xe-133m	2.91E+03	1.31E+03
Xe-133	9.84E+04	4.36E+04
Xe-135m	1.36E+03	5.93E+02
Xe-135	2.56E+04	1.16E+04

TABLE 15.7-13

REFUELING ACCIDENTS
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Design Basis Accident)

Isotope	Activity Released to Environment (curies)	
	Equipment Handling Accident (455 Failed Rods)	Fuel Handling Accident (191 Failed Rods)
I-131	5.12E+01	2.28E+01
I-132	3.84E+01	1.71E+01
I-133	3.21E+01	1.43E+01
I-134	0.00E+00	0.00E+00
I-135	5.23E+00	2.33E+00
Kr-83m	1.81E+01	7.88E+00
Kr-85m	2.80E+02	1.22E+02
Kr-85	1.58E+03	6.68E+02
Kr-87	3.18E-02	1.42E-02
Kr-88	7.63E+01	3.45E+01
Xe-131m	5.56E+02	2.74E+02
Xe-133m	2.89E+03	1.30E+03
Xe-133	9.82E+04	4.35E+04
Xe-135m	2.35E+02	1.02E+02
Xe-135	2.37E+04	1.08E+04

TABLE 15.7-14

REFUELING ACCIDENTS
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Realistic Accident)

Isotope	Reactor Building Airborne Activity (curies)	
	Equipment Handling Accident (455 Failed Rods)	Fuel Handling Accident (191 Failed Rods)
I-131	2.21E+01	9.80E+00
I-132	2.80E+01	1.25E+01
I-133	2.23E+01	9.94E+00
I-134	1.22E-06	5.49E-07
I-135	3.68E+00	1.64E+00
Kr-83m	9.32E+00	4.08E+00
Kr-85m	1.17E+02	5.13E+01
Kr-85	2.82E+02	1.20E+02
Kr-87	1.90E-02	8.51E-03
Kr-88	3.47E+01	1.57E+01
Xe-131m	1.98E+02	9.80E+01
Xe-133m	1.05E+03	4.71E+02
Xe-133	3.54E+04	1.57E+04
Xe-135m	4.91E+02	2.14E+02
Xe-135	9.21E+03	4.17E+03

TABLE 15.7-15

REFUELING ACCIDENTS
AIRBORNE RELEASED TO ENVIRONMENT (curies)
(Realistic Accident)

Isotope	Activity Released to Environs (curies)	
	Equipment Handling Accident (455 Failed Rods)	Fuel Handling Accident (191 Failed Rods)
I-131	2.05E+00	9.11E-01
I-132	2.46E+00	1.10E+00
I-133	2.06E+00	9.18E-01
I-134	0.00E+00	0.00E+00
I-135	3.35E-01	1.49E-01
Kr-83m	6.50E+00	2.85E+00
Kr-85m	1.01E+02	4.41E+01
Kr-85	2.83E+02	1.20E+02
Kr-87	1.15E-02	5.13E-03
Kr-88	2.74E+01	1.24E+01
Xe-131m	1.99E+02	9.82E+01
Xe-133m	1.04E+03	4.66E+02
Xe-133	3.53E+04	1.57E+04
Xe-135m	8.48E+01	3.70E+01
Xe-135	8.56E+03	3.87E+03

TABLE 15.7-16

REFUELING ACCIDENTS – RADIOLOGICAL EFFECTS

	Doses (REM TEDE)	
Design Basis Analysis	Equipment Handling Accident (455 Failed Rods)	Fuel Handling Accident (191 Failed Rods)
Acceptance Criterion - Offsite	6.30	6.30
2 HR Exclusion Area Boundary	1.97	0.882
Low Population Zone (30 day)	0.116	0.0521
Acceptance Criterion - CRHE	5.00	5.00
CRHE	0.1636	0.0731
Realistic Analysis	Equipment Handling Accident (455 Failed Rods)	Fuel Handling Accident (191 Failed Rods)
Acceptance Criterion - Offsite	6.30	6.30
2 HR Exclusion Area Boundary	0.083	0.037
Low Population Zone (30 day)	0.0031	0.0014
Acceptance Criterion - CRHE	5.00	5.00
CRHE	0.031	0.014

CRHE - Control Room Habitability Envelope

TABLE 15.7-17

REFUELING ACCIDENTS - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

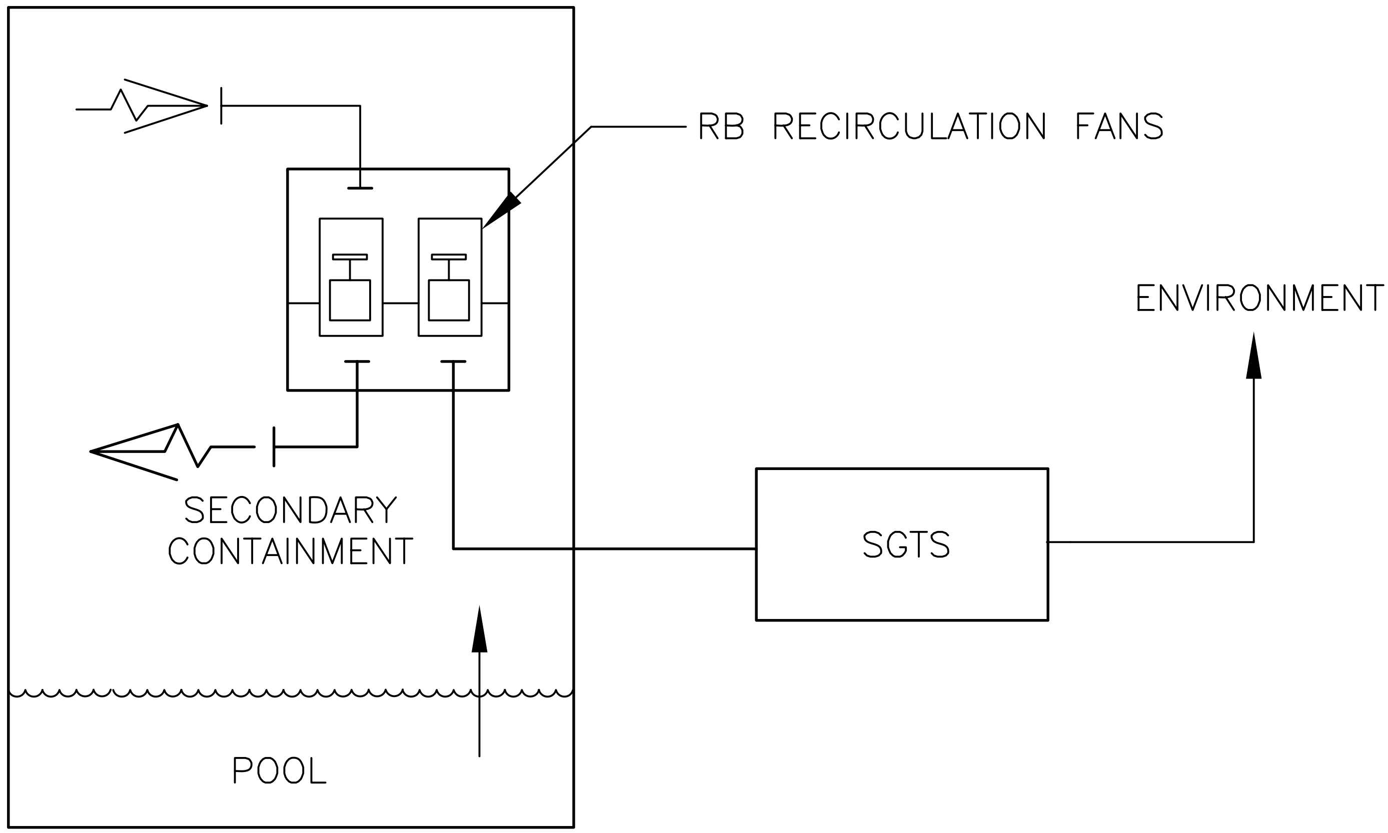
	Design Basis Assumptions	Realistic Assumptions
I. Data and assumptions used to estimate radioactive sources from postulated accidents		
A. Reactor power (MWt)	4032	4032
B. Radial peaking factor	1.6	1.6
C. Fuel damaged EHA FHA	 455 191	 455 191
D. Release of activity by nuclide	5 percent of noble gases and halogens 10 percent of Kr-85 8 percent of I-131 12 percent of alkali metals	1.8 percent of noble gases 0.32 percent of iodines
II. Data and assumptions used to estimate activity released		
A. Secondary containment leak rate	All activity released to environment over 2 hour period	All activity released to environment over 2 hour period
B. SGTS filtration efficiencies (percent) iodines noble gases	 0% all iodine species for the first 10 minutes, then 99% thereafter for the duration of the event 0	 0% all iodine species for the first 10 minutes, then 99% thereafter for the duration of the event 0
C. Fuel pool noble gase decontamination factor	0	0
D. Fuel pool iodine decontamination factor	138	138
E. Decay time prior to accident, hr.	24	24
F. Time delay in SGTS filtration (min)	10	10

III. Data And Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume (ft^3)	518,000	518,000
B. Control room free volume (ft^3)	110,000	110,000
C. Control structure filtered air intake flow(cfm)	5229 – 6391	5229 - 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	600	600
F. Control structure filter efficiency for iodine (percent)	99	99
IV Dispersion data		
A. Boundary and LPZ distance (meters)	549/4827	549/4827
B. X/Q's for EAB C. X/Q's for LPZ D. X/Q's for CRHE	Table 2.3-92 (0.5 percentile) Table 2.3-105(0.5 percentile) Appendix 15B	Table 2.3-92 (50 percentile) Table 2.3-105 (50 percentile) Appendix 15B
V. Dose data		
A. Method of dose calculation	Reg. Guide 1.183 Appendix 15B	Appendix 15B
B. Dose conversion	Appendix 15B	Appendix 15B
C. Activity in secondary containment	Table 15.7-12	Table 15.7-14
D. Activity released to environment	Table 15.7-13	Table 15.7-15
E. Doses	Table 15.7-16	Table 15.7-16

TABLE 15.7-18

SEQUENCE OF EVENTS FOR SJAE FAILURE

Approximate Elapsed Time	Events
0 sec.	Event begins with system failure and the release of radioactive noble gases and iodines to the building.
0 to 24 hours	Event detection and termination of release: Event detection is based on a loss of offgas system flow. Notification of control room of loss of offgas system flow. Verification of loss of offgas system flow by one or more of the following: 1. Verification of an activity release into the Turbine and/or Radwaste Building or to the environment by evaluation of appropriate ARMs, CAMs, Turbine Building Vent SPINGs, and/or airborne radiation surveys. 2. Verification of loss of offgas system flow by evaluating the operability of the offgas system flow instrumentation and monitoring of other offgas system parameters. Operator actions begin with: 1. Initiation of appropriate system isolations. 2. Manual scram actuation. 3. Assurance of reactor shutdown cooling.



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LEAKAGE PATH
FOR
REFUELING ACCIDENTS

FIGURE 15.7-1, Rev 54

AutoCAD: Figure Fsar 15_7_1.dwg