15. ACCIDENT ANALYSES

15.0 TRANSIENT ANALYSES

This chapter presents an analytical evaluation of the response of the plant to postulated malfunctions or failures of equipment. These incidents are postulated and their consequences analyzed despite the many precautions which are taken in the design, construction, quality assurance, and plant operation to prevent their occurrence. The potential consequences of such occurrences are then examined to determine their effect on the plant, to determine whether plant design is adequate to minimize consequences and to assure that the health and safety of the public and plant personnel are protected from the consequences of even the most severe of the hypothetical incidents analyzed.

The structure of this section is based on the eight by three matrix specified in Reference 1. Initiating events are placed in one of eight categories of process variable perturbation specified in Reference 1 and are discussed in Section 15.0.1. The frequency of each incident[1] was estimated, and each incident was placed in one of three frequency categories specified in Reference 1 and discussed in Section 15.0.1.

15.0.1 IDENTIFICATION OF CAUSES AND FREQUENCY CLASSIFICATION

15.0.1.1 Safety Analyses Applicable after Permanent Cessation of Power Operation

Per References 2 and 3, SONGS has permanently ceased operation and removed all nuclear fuel from both units reactor vessels. The irradiated fuel will be stored in the spent fuel pool (SFP) and in the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped offsite. In this configuration, the SFP and its systems are dedicated only to spent fuel storage and handling. In this condition, the number of credible accidents/transients is significantly smaller than for a plant authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Accidents/transients that are no longer applicable in a permanently defueled condition have been deleted from this chapter, where appropriate. With all irradiated fuel being stored in the SFP and the ISFSI, the reactor, Reactor Coolant System (RCS) and secondary systems are no longer in operation and have no function related to storage and handling of irradiated fuel. With the permanent cessation of power operation and the permanent removal of the fuel from the reactor vessel, the accident/transient initial conditions/initial reactor power level of the reactor core cannot be achieved and, as such, most of the accident/transient scenarios are not possible. Therefore, the postulated UFSAR Chapter 15 accidents/transients involving failure or malfunction of the reactor, RCS, or secondary systems are no longer applicable.

[1] Incidents are defined in this section as the initiating event.

The initiating events for which analyses are presented are listed in Table 15.0-1 along with their respective section designations.

Certain initiating events which are suggested for consideration in Reference 1 have not been explicitly analyzed. These initiating events, along with the reasons for omission of their analyses, are provided in the appropriate paragraphs in this chapter.

The frequency of each incident has been estimated and each incident is placed in one of the frequency categories. These frequency categories are defined as follows:

A. Moderate Frequency Incidents

   These are incidents, any one of which may occur during a calendar year for a particular plant.

B. Infrequent Incidents

   These are incidents, any one of which may occur during the lifetime of a particular plant.

C. Limiting Faults

   These are incidents that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These events also bound all moderate frequency and infrequent events of similar nature.

15.0.2 SYSTEMS OPERATION

In the permanently defueled condition no automatic operation of any system is credited in mitigating the consequences of an incident.

Table 15.0-1
CHAPTER 15 INITIATING EVENTS

Paragraph	Event
Moderate Frequency Incidents	
	NONE
Infrequent Incidents	
	NONE
Limiting Faults	
15.1.1.1	Radioactive waste gas system leak or failure
15.1.1.2	Radioactive liquid waste system leak or failure (gas release to atmosphere)
15.1.1.3	Postulated radioactive releases due to liquid tank failures
15.1.1.4	Design Basis fuel handling accident inside fuel building
15.1.1.5	Spent fuel cask drop accidents
15.1.1.6	Spent fuel pool gate drop accident
15.1.1.7	Test equipment drop
15.1.1.8	Spent fuel pool boiling accident
15.1.1.9	Spent fuel assembly drop
15.1.1.10	Use of miscellaneous equipment under 2000 lbs

15.0.3 RADIOLOGICAL CONSEQUENCES

This subsection summarizes the assumptions, parameters, and calculational methods used to determine the doses that result from postulated accidents.

San Onofre Units 2 and 3 are licensed for full implementation of the Alternative Source Term (AST) methodology for radiological consequence analyses. All radiological analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the Total Effective Dose Equivalent (TEDE) criteria of 10CFR50.67.

Appendix 15G identifies the models used to calculate offsite radiological doses due to postulated accidents evaluated in accordance with the AST dose analysis methodology of Regulatory Guide 1.183.

The definition of a limiting fault, as provided in Subsection 15.0.1, is an incident that is not expected to occur but is postulated because its consequences include the potential for the release of significant amounts of radioactive materials. For the design basis case, very conservative assumptions are made regarding the event parameters. The parameters that have been modified for the realistic analyses are presented in the description of each limiting fault.

Information used repetitively throughout the section is provided in Appendix 15G for AST radiological calculations. This appendix contains information on dose models, atmospheric dispersion factors, and activity release models.

15.0.4 REFERENCES

1. NRC Regulatory Guide 1.70, Revision-2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," September 1975.

2. Letter from Peter T. Dietrich to U.S. Nuclear Regulatory Commission, “Docket No. 50-361 Permanent Removal of Fuel from the Reactor Vessel San Onofre Nuclear Generating Station Unit 2.”, dated July 22, 2013.

3. Letter from Peter T. Dietrich to U.S. Nuclear Regulatory Commission, “Docket No. 50-362 Permanent Removal of Fuel from the Reactor Vessel San Onofre Nuclear Generating Station Unit 3.”, dated June 28, 2013.

15.1 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

15.1.1 LIMITING FAULTS

15.1.1.1 Radioactive Waste Gas System Leak or Failure

The evaluation of the radiological consequences for a Radioactive Waste Gas System leak or failure assumes a minimum of 17 months since the shutdown of Units 2 and 3.

This event is modeled with the Alternative Source Term (AST). Additional assumptions associated with AST modeling are provided in Appendix 15G.

15.1.1.1.1 Identification of Causes and Frequency Classification

The most limiting waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive fission gases that were previously stored in one waste gas decay tank. The gaseous radwaste system (GRS) is described in Chapter 11.

This accident is considered a limiting fault, and a rupture of a waste gas decay tank is analyzed to define the worst consequences of a gaseous release that could result from any malfunction in the gaseous radwaste system.

15.1.1.1.2 Sequence of Events and System Performance

15.1.1.1.2.1 Sequence of Events and System Performance

It is assumed that the plant had been operating at 3560 MWt (105% of the originally licensed power level of 3390 MWt) with 1% failed fuel for an extended period sufficient to achieve equilibrium radioactive concentrations in the reactor coolant. The maximum gas activity release from either plant would have occurred after shutdown and coolant degasification. The gases from one reactor coolant inventory in a single tank would provide an upper limit for stored gas activity. This tank is assumed to rupture and all of the fission gases are assumed to be released to the atmosphere in a 2-hour period. Tables 15.1-1 and 15.1-2 list the conservative assumptions for waste gas decay tank rupture event and waste gas decay tank inventory prior to release, respectively.

15.1.1.1.3 Radiological Consequences

15.1.1.1.3.1 Assumptions and Calculational Model

The χ/Q value (5% level) used is representative of the meteorology for the 0- to 2-hour interval at the location of the dose point; i.e., at the actual site boundary and at the outer boundary of the LPZ. The leak rate from the auxiliary building is such that the total leakage is equal to the total release of activity from the tank. The Alternative Source Term (AST) models used to calculate doses are discussed in Appendix 15G.

Table 15.1-1
ASSUMPTIONS FOR WASTE GAS DECAY TANK RELEASE ACCIDENT

	Assumption
Source Data	Power level prior to accident is 3,560 MWt.
	RCS radioactive concentrations are maximum values based on 1% failed fuel.
	All gases stripped from processing a RCS volume are immediately passed to the gas decay tank which fails. Gas stripper partition factor is 1 for noble gases, 10^{-3} for iodines
	A decontamination factor of 10 is assumed for the CVCS purification ion-exchanger for iodine.
	Assume 17 month decay.
	No credit taken for radioactive decay during transit.
	Tank activity is presented in Table 15.1-2
Activity Release	All gases released from tank leak from auxiliary building at ground level within a 2-hour period.
Meteorological Data	5% level χ/Qs per Appendix 15G.
Dose Data	Doses calculated using the model discussed in Appendix 15G.

Table 15.1-2
RADIOLOGICAL RELEASES AS A RESULT OF A WASTE
GAS DECAY TANK RELEASE ACCIDENT

Isotopes	Activity Release to Atmosphere* (Ci)
I-131	9.4615×10^{-21}
I-132	0.0
I-133	0.0
I-134	0.0
I-135	0.0
Kr-85m	0.0
Kr-85	1.1823×10^{3}
Kr-87	0.0
Kr-88	0.0
Xe-131m	0.0
Xe-133	0.0
Xe-135	0.0
Xe-135m	0.0
Xe-138	0.0

*Assumes 17 month decay

15.1.1.1.3.2 Results and Conclusions

The results of a postulated waste gas decay tank rupture are presented in Table 15.1-2A. The doses at the exclusion area boundary (EAB) and low population zone (LPZ) are less than the 100 mRem TEDE offsite dose criterion per Regulatory Issue Summary 2006-04.

Table 15.1-2A
RADIOLOGICAL EXPOSURES AS A RESULT OF A WASTE
GAS DECAY TANK RELEASE ACCIDENT

DOSE RECEPTOR	DOSE (mRem TEDE)	ACCEPTANCE CRITERION (mRem TEDE)
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	0.14	100
LPZ (30-day accident duration)	0.00	100

15.1.1.2 Radioactive Liquid Waste System Leak or Failure (Gas Release to Atmosphere)

The evaluation of the radiological consequences for a Liquid Radioactive Waste System leak or failure (with release to atmosphere) conservatively does not assume any post-shutdown decay time. The doses would be less if a decay time was assumed.

This event is modeled with the Alternative Source Term (AST), per Appendix 15G.

15.1.1.2.1 Identification of Causes and Frequency Classification

Liquid releases considered include rupture of radwaste tanks, refueling water storage tanks, primary ion-exchangers, and the blowdown demineralizer neutralization sump line. The most limiting of these is defined as an unexpected and uncontrolled release of the radioactive liquid stored in a radwaste secondary tank. Rupture of these tanks is considered a limiting fault. A radwaste secondary tank rupture would release the liquid contents in the auxiliary building (radwaste area). Refer to Section 15.1.1.3 for evaluation of accidental release of radioactive liquid with respect to 10 CFR 20 limits.

The radiological consequences of the release to the atmosphere of radioactive fission gases are considered in this evaluation.

15.1.1.2.2 Sequence of Events and System Operation

15.1.1.2.2.1 Design Basis Sequence of Events and System Operation

It is assumed that radwaste secondary tank activity is based on 1% failed fuel. The activity is consistent with the maximum activity condition at full power plant operation.. Design basis assumptions are presented in Table 15.1-3. Source terms are shown in Table 15.1-4.

A radwaste secondary tank is assumed to rupture, releasing the contents of the tank to the auxiliary building. All of the radioactive fission gases are assumed to be released to the outside atmosphere in 2 hours.

15.1.1.2.3 Radiological Consequences

The assumptions used to evaluate the rupture of a radwaste secondary tank are listed in Table 15.1-3 and the radioactive inventory in the tanks is listed in Table 15.1-4. The Alternative Source Term (AST) models used to calculate doses are discussed in Appendix 15G.

Offsite doses due to the rupture of a radwaste secondary tank are presented in Table 15.1-4A. As shown, they are less than the 100 mRem TEDE offsite dose criterion per Regulatory Issue Summary 2006-04.

Table 15.1-3
ASSUMPTIONS FOR LIQUID TANK RUPTURE (RELEASE TO ATMOSPHERE)

	Design Basis Assumption
Source Data	RCS radioactive concentrations are maximum values based on 1% failed fuel and 3560 MWt (105% of the originally licensed power level of 3390 MWt).
	Tank activity presented in Table 15.1-4.
	Iodine partition factor after tank failure (1.0).
Activity Release	Gases and iodines are released from auxiliary building at ground level within a 2-hour period.
Meteorological Data	5% level χ/Qs per Appendix 15G.
Dose Data	Doses calculated using the model discussed in Appendix 15G.

Table 15.1-4
RADIOLOGICAL RELEASE AS A RESULT OF LIQUID TANK RUPTURE
(RELEASE TO ATMOSPHERE)

	Radioactivity Released (Ci)
Isotopes	Assumptions
I-131	2.010×10^{-1}
I-132	2.875×10^{-3}
I-133	1.662×10^{-1}
I-134	1.868×10^{-4}
I-135	3.024×10^{-2}
Kr-85m	1.570×10^{-1}
Kr-85	2.33×10^{0}
Kr-87	8.506×10^{-3}
Kr-88	1.281×10^{-1}
Xe-131m	1.047×10^{0}
Xe-133	1.399×10^{2}
Xe-135m	3.171×10^{-4}
Xe-135	1.501×10^{0}
Xe-138	1.316×10^{-4}

Table 15.1-4A
RADIOLOGICAL EXPOSURES AS A RESULT OF LIQUID TANK RUPTURE
(RELEASE TO ATMOSPHERE)

DOSE RECEPTOR	DOSE (mRem TEDE)	ACCEPTANCE CRITERION (mRem TEDE)
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	7.1	100
LPZ (30-day accident duration)	1.4	100

15.1.1.3 <u>Postulated Radioactive Release Due to Liquid Tank Failures</u>

15.1.1.3.1 Identification of Causes and Frequency Classification

Accidents involving release of radioactive liquids from tanks may involve rupture of tanks inside the containment, inside the auxiliary building, or of the refueling water or condensate storage tanks located outside. Tanks inside the containment include the reactor coolant drain tank and quench tank, which are designed to Seismic Category II criteria. The volume control tank and all liquid radwaste processing tanks, located in the Seismic Category I auxiliary building, are designed to Seismic Category I and II criteria, respectively. The Seismic Category I refueling water and Seismic Category II condensate storage tanks located in the yard area are surrounded by retention basins. As described in Chapter 11, the Condensate Storage Tank is administratively controlled to ensure that any overflow will be within 10 CFR 20 limits. An accident involving a liquid tank failure is considered a limiting fault.

15.1.1.3.2 Sequence of Events and System Operations

A hypothetical rupture of a tank inside the containment would release radioactive liquid to the containment sump where it would be collected and processed through the radioactive waste disposal system. The containment has a steel-lined interior structure; therefore, there is no pathway for leaked fluids to affect water in unrestricted areas.

Radioactive tanks in the auxiliary buildings are contained in separate, concrete-walled rooms. These rooms are provided with water stops at the construction joints and seals wherever piping penetrates through the concrete walls to the tanks. Drain lines from the rooms are routed to the radwaste area sump. Spilled leakage would be collected in the sump and may be stored in tanks or processed through the radioactive waste disposal system. Radioactive liquids released from a RWST or the condensate storage tank would be contained in the concrete retention basins surrounding each tank. The condensate storage tank is subject to administrative controls described in Chapter 2 for outdoor, unprotected tanks that ensure any uncontrolled release of tank contents would be within 10 CFR 20 limits.

The liquid waste disposal system is designed to minimize or preclude discharge of plant-originated radioactive liquid wastes to the surrounding environment. However, the system has optional capability of discharge to the discharge conduits within the limits of 10CFR20. As discussed in Chapter 2, the radioactive waste discharge line is the only release path for radioactive effluent discharges into the surface water in an unrestricted area. Chapter 11 discusses the administrative controls and automatic interlocks, together with the fail-safe design of the instrumentation and control devices, which provide assurance against unauthorized or excessive releases of radioactive liquids.

15.1.1.3.3 Radiological Consequences

No credible accident scenarios exist that would exceed 10 CFR 20 limits. Therefore, no formal radiological consequence evaluation of an accident is warranted.

Refer to Chapter 2 for a discussion of the effects of a postulated radioactive liquid tank failure on surface water and groundwater.

15.1.1.4 <u>Design Basis Fuel Handling Accident Inside Fuel Building</u>

15.1.1.4.1 Identification of Causes and Frequency Classification

The possibility of a fuel handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel handling operations (refer to Chapter 9). All fuel handling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety and fuel handling.

Design of the fuel storage racks and handling facilities in the fuel storage area is such that fuel will always be in a subcritical geometrical array without crediting the boron concentration in the fuel pool water. Refer to Chapter 9 for a discussion of spent fuel racks. The spent fuel pool and pool water contains boron in accordance with Technical Specification 3.1.2, “Fuel Storage Pool Boron Concentration.” Natural convection of the surrounding water provides adequate cooling of fuel during handling and storage. Adequate cooling of the water is provided by forced circulation in the spent fuel pool cooling system. At no time is a fuel assembly removed from the water. Fuel failure during fuel handling, as a result of inadvertent criticality or overheating, is not possible.

In the fuel building, a fuel assembly could be dropped in the fuel transfer canal or in the spent fuel pool. In addition to the area radiation monitor located in the spent fuel cask area, portable radiation monitors capable of emitting audible alarms are located in this area during fuel handling operations. Should a fuel assembly be dropped in the spent fuel pool (including cask pool and transfer pool) releasing radioactivity above a prescribed level, the airborne radiation monitors sound an alarm, alerting personnel. Interlocks and mechanical stops prevent the spent fuel cask handling crane from moving the cask over stored irradiated fuel and limit cask

movement (refer to Chapter 9). The probability of a fuel handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility as previously mentioned. However, since the fuel handling accident is considered a limiting fault, it is postulated that a fuel assembly is dropped during fuel handling operations in the fuel building, breaching the cladding of the fuel pins and releasing fission products contained in the gap region of the fuel pin.

15.1.1.4.2 Sequence of Events and System Operation

15.1.1.4.2.1 <u>Design Basis Sequence of Events and System Operation</u>

A description of the fuel handling procedure appears in Chapter 9.

For the design basis accident, the failure of 472 fuel rods was evaluated. The failure of 472 fuel rods is the largest number of fuel rods that could fail from the assembly drop as described in Section 15.1.1.4.2.2.

The resultant release of radioactivity, after escaping from the spent fuel pool, is conservatively assumed to be exhausted from the fuel handling building during a 2 hour period. See Table 15.1-5 for parameters used in evaluating this event.

15.1.1.4.2.2 <u>Structural Evaluation of Fuel Assembly</u>

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

The fuel assemblies are stored within the spent fuel rack at the bottom of the spent fuel pool. The top of the rack extends 13.2 inches above the tops of the stored fuel assemblies. A dropped fuel assembly could not strike more than one fuel assembly in the storage rack. Impact could occur only between the ends of the involved fuel assemblies, the bottom end fitting of the dropped fuel assembly impacting against the top end fitting of the stored fuel assembly. The maximum drop distance for this event is 74 inches from the bottom of a fuel assembly residing in the spent fuel handling machine to the top of a fuel assembly in the spent fuel storage racks. For the 74-inch drop, the fuel assembly impact velocity is 215 in./s and the impact stress in the fuel rod cladding is 20,100 psi. Criticality is not a concern for this postulated event, since the rack configuration remains intact.

Two cases were considered for the accidental drop of a fuel assembly onto or into the racks. These were:

A. Westinghouse 14 x 14 standard fuel assembly with control rods, total dry weight of 1260 pounds, dropped from a conservative height of 24.9 feet above the pool floor.

B. Combustion Engineering 16 x 16 fuel assembly with control rods, total dry weight of 1540 pounds, dropped from a conservative height of 21.17 feet above the pool floor.

The drop orientations considered were a drop of an assembly onto the top of the racks with the assembly in a vertical position, drop of an assembly onto the top of the racks with the assembly in an inclined position, and a drop of a fuel assembly through an empty cell to the bottom of the pool.

The results of these analyses show that with 1800 ppm boron in the fuel pool water, fuel criticality does not occur. Thus, the acceptance criterion of no fuel criticality is met for all credible fuel drop accidents. Further, each of these three drop orientations was evaluated to determine the velocity of impact with the pool liner. In each case the structure at the lower end of the assembly had enough strain energy capacity to absorb the drop kinetic energy. When consideration was given to the "footprint" of the dropped assembly, the stresses imposed on the pool liner were determined not to perforate the pool liner for any of the drop accidents.

The maximum possible drop distance for a fuel assembly in the spent fuel pool is 254 inches. This is the distance from the bottom of a fuel assembly in the spent fuel handling machine to the spent fuel pool floor. For this worst case drop, the velocity of the fuel assembly at impact with the fuel pool floor is 362 in./s and the impact stress in the fuel rod cladding is 34,000 psi.

The analyses of the fuel assembly vertical drops reported above were performed with a calculational model that incorporates skin friction and form drag of the fuel assembly into a mathematical formulation of the fuel assembly motion which is given below:

$$\ddot{\chi} + [(F_S + F_D)/M]\ \dot{x}^2 \quad - W/M = 0 \qquad (1)$$

where:

F_S = skin friction coefficient

F_D = form drag coefficient

M = mass of a fuel assembly

W = net weight of a fuel assembly

$\dot{x}$ = net velocity

$\ddot{\chi}$ = acceleration

The equation employed in calculating the impact stresses in the fuel rod clad is as follows:

$$\sigma_1 = \dot{X}_I \sqrt{E\rho} \qquad (2)$$

where:

σ_1 = impact stress

$\dot{X}_I$ = impact velocity

E = modulus of elasticity

ρ = mass density

The allowable stress in the fuel rod cladding, σ yield is 49,000 psi. This is the minimum yield stress value for unirradiated Zircaloy-4 and is conservative for irradiated fuel. Thus, for the worst case fuel assembly vertical drop, the impact stresses which result from absorbing the kinetic energy of the drop are below the yield stress of the clad and no fuel rod failures will occur.

The original design basis structural analysis postulated that the worst case fuel assembly horizontal impact results from a vertical drop of the maximum possible distance (254 inches) to the fuel pool floor, followed by rotation of the fuel assembly to the horizontal position. During this rotation, it is postulated that the assembly strikes a protruding structure. The fuel storage pool is designed without such protruding structures and hence the shape and nature of the assumed member is indeterminate. For this analysis, therefore, a line load has been assumed for the most severe accident.

The original design basis structural analysis of this fuel assembly drop has revealed that the most severe impact location is between the top two spacer grids due to the relatively higher impact velocity of the top of the fuel assembly. Since the impact area is within the fuel rod upper plenum region, the fuel pellets do not provide clad support and do not enter into the failure analysis. To obtain an estimate of the number of fuel rods which might fail, the fuel assembly was modeled and calculations performed with the SHOCK[1] computer code. The SHOCK code allows modeling of the fuel assembly to include consideration of localized deformations about the impact point as well as general bending of the fuel assembly. The code's input data describing fuel material properties and pool conditions were kept consistent with the circumstances of the accident; i.e., irradiated fuel assembly material properties, water and fuel rod cladding temperatures corresponding to spent fuel pool conditions. For this worst case fuel

assembly drop accident, no more than four rows of fuel rods (60 rods) would fail due to the combined bending and localized deformation which results from absorbing the kinetic energy at impact. For conservatism, fuel rod cladding failure was assumed to occur if the stress distribution across the fuel rod tube reached a uniform value equal to the yield stress of irradiated Zircaloy.

The current structural evaluation was originated to determine the extent of fuel rod damage produced by a fuel assembly (i.e., fuel bundle) being dropped from the fuel handling device and impacting one or more fuel bundles in the spent fuel rack during fuel handling operations. The structural evaluation addresses increases in the fuel bundle weight and to include the weights of components, handling grapples, and discretionary margin. Fuel rod damage is limited to 236 rods per bundle (472 pins when two bundles are considered, dropped and impacted) regardless of the type(s) or number(s) of impact.

In the current structural evaluation, energy balance theory is employed to determine the number of damaged fuel rods resulting from the postulated events. The methodology used in the current structural evaluation is in keeping with the original structural analysis.

Due to the design of the spent fuel rack, each spent fuel bundle is placed in a separate rack with very small gaps between the full assembly and the rack. Moreover, the height of the rack exceeds the height of the spent fuel bundle. Therefore, the spent fuel bundle may be impacted only by a vertically falling fuel assembly hitting the bundle symmetrically, i.e. the axis of the dropped fuel assembly must coincide with the axis of the impacted bundle (asymmetrical contact is practically non-achievable, and a horizontally dropped assembly cannot hit a fuel bundle). No more than one impacted fuel bundle may be affected, and no tipping of the impacted fuel bundle is achievable. Therefore, the only loading on an impacted fuel bundle in the spent fuel rack is the result of a symmetrical impact with the vertically dropped fuel bundle. In addition, the structural evaluation considers that the vertically dropped fuel bundle could tip over after impact with the spent fuel rack and come to rest with a horizontal impact.

For the fuel handling accident inside the fuel handling building, the current structural evaluation for the bundle drop scenario at the spent fuel rack location determines that a maximum of 472 fuel rods will fail as a result of a vertical drop of the fuel assembly for a dropped weight up to 2065 pounds. The drop weight of 2065 pounds represents a bundle dry weight of 1495 pounds, plus 120 pounds of components (e.g., control element assembly [CEA], neutron sources, etc.) plus 400 pounds of grapples, plus 50 pounds discretionary margin.

15.1.1.4.3 Radiological Consequences

The assumptions used to evaluate the fuel handling accident are provided in Regulatory Guide 1.183 Appendix B. Analysis input values are listed in Table 15.1-5. The Alternative Source Term (AST) models used to calculate doses are discussed in Appendix 15G.

Offsite doses due to FHA-FHB are presented in Table 15.1-6. As shown, they are less than the 6.3 Rem TEDE offsite dose criterion per Table 6 of Regulatory Guide 1.183.

Table 15.1-5
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING

FHA-FHB PARAMETER	MODELED VALUE
Dose acceptance criteria, Rem TEDE	
EAB	6.3
LPZ	6.3
FHA-FHB source term	
Maximum decay time after reactor shutdown, hours	12,240 (17 months)
Average fuel rod isotope inventory at 12,240 hours, curies/rod	per Appendix 15G
Radial peaking factor applied to all failed fuel rods	1.75
Number of failed fuel rods	472
Core fission product fractions in fuel rod gaps	
Iodine-131	0.08
Krypton-85	0.10
Other noble gases (Krypton, Xenon)	0.05
Other Halogens (Iodine, Bromine)	0.05
Alkali Metals (Cesium, Rubidium)	0.12
Fraction of gap activity released to the fuel storage pool	1.00
Minimum water depth above damaged fuel rods, feet	23
Fuel storage pool decontamination factors	
Iodines (effective DF)	200
Noble Gases	1
Particulates	Infinite
Iodine composition above the fuel storage pool, percent of iodine	
Elemental iodine	57
Organic iodide	43
Fuel Handling Building model	
Unfiltered activity release duration from FHB, hours	2
FHB net free volume, cubic feet	365,305
FHB air exhaust flow rate, ft^3/minute	22,000
Offsite dose evaluation model	per Appendix 15G

Table 15.1-6
FHA-FHB DOSE CONSEQUENCES

DOSE RECEPTOR	FHA-FHB DOSE (REM TEDE)	ACCEPTANCE CRITERION (REM TEDE)
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	0.20E-3 (0.20 mRem TEDE)	6.3
LPZ (30-day accident duration)	0.01E-3 (0.01 mRem TEDE)	6.3

15.1.1.5 Spent Fuel Cask Drop Accidents

This section analyzes spent fuel cask drop events. Three situations are considered: a spent fuel cask drop into the spent fuel pool, a spent fuel cask dropped by the Cask Handling Crane onto a flat surface, and a spent fuel transfer cask drop (due to a seismic event) from the upper shelf in the cask pool back into the lower portion of the cask pool. The spent fuel transfer cask may be loaded with up to 32 fuel assemblies.

The spent fuel cask drop events are evaluated based on the ability of the cask drops to cause the release of radioactive materials. This includes consideration of the allowed travel paths of the casks, their lift heights, and the items onto which they can be dropped.

15.1.1.5.1 Cask Drop Into Spent Fuel Pool

As discussed in Chapter 9, the cask handling crane is prohibited from traveling over the spent fuel pool or any unprotected safety-related equipment. Thus, an accident resulting from dropping a cask or other major load into the spent fuel pool is not credible. In addition, single-failure-proof cranes will be used at Units 2 and 3 to lift a spent fuel transfer cask out of a cask pool.

15.1.1.5.2 Cask Drop to Flat Surface

As discussed in Chapter 9, the potential drop of a spent fuel cask is limited to less than an equivalent 30-foot drop onto a flat, essentially unyielding, horizontal surface. The only fuel that can be damaged is inside the sealed spent fuel cask making this event less limiting than a “Cask Drop from Upper Shelf in the Cask Pool (Section 15.1.1.5.3).” Thus, the radiological consequences of this accident are not evaluated. In addition, single-failure-proof cranes will be used at Units 2 and 3 to lift a spent fuel transfer cask out of a cask pool.

15.1.1.5.3 Cask Drop from Upper Shelf in the Cask Pool

Even though single-failure-proof cranes will be used at Units 2 and 3 to lift a spent fuel transfer cask out of a cask pool, a drop can be postulated when the cask is placed on the upper shelf (i.e., step) of a cask pool for lifting yoke change-out, prior to the transfer cask being welded closed. During this evolution, the transfer cask is not restrained and could fall back into the lower portion of the cask pool if an earthquake occurs.

It is assumed that a minimum of 17 months have elapsed since permanent discharge from the core for Unit 2 or 3 fuel assemblies that are loaded into a transfer cask. The fuel rods from all 32 fuel assemblies that may be present in a transfer cask are conservatively assumed to rupture on impact with the bottom of the cask pool. All of the radioactive iodine and noble gases present in the gap volumes of the decayed fuel rods are assumed to be released from the unwelded transfer cask.

Other than the number of fuel assemblies considered to fail, the cask drop accident is modeled identically to that of the fuel handling accident in the fuel handling building (FHA-FHB), as addressed in UFSAR Section 15.1.1.4.

The release of radioactive material to the atmosphere represents a potential exposure hazard to the general public at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ). The EAB, and LPZ doses are calculated using the dose evaluation model described in Section 15.1.1.4 and Appendix 15G. Consistent with the FHA-FHB event, the cask drop accident radiological criterion is 6.3 Rem TEDE for the EAB and LPZ doses. The resulting cask drop accident offsite doses are listed in Table 15.1-6A. The analysis demonstrates that the cask drop accident criteria are met.

Table 15.1-6A
CASK DROP ACCIDENT DOSE CONSEQUENCES

DOSE RECEPTOR	CASK DROP DOSE (REM TEDE)	ACCEPTANCE CRITERION (REM TEDE)
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	3.09E-3 (3.09 mRem TEDE)	6.3
LPZ (30-day accident duration)	0.09E-3 (0.09 mRem TEDE)	6.3

15.1.1.6 Spent Fuel Pool Gate Drop Accident

The spent fuel pool gates consist of the transfer pool gate and the cask pool gate. During normal opening and closing, the gates operate on rails and are not subject to being dropped. The gates are periodically removed from the rails to perform maintenance on the gate seals. During removal and reinstallation, the gates are temporarily moved over the high density spent fuel storage racks. A gate drop accident is postulated where a gate is accidentally dropped while being carried over the racks.

To eliminate the potential for adverse radiological consequences, the following administrative controls provide assurance that the cask pool and transfer pool gates will not impact fuel assemblies stored in the impact zone or while reconstitution activities are in progress:

A. The spent fuel pool gates shall not be removed from the installed position while reconstitution activities are in progress. Normal operation of opening and closing the gates is permitted.

B. Prior to and during rigging for removal and reinstallation of the cask pool and transfer pool gates, all fuel assemblies shall be located outside the potential primary impact zones:

   1. The primary impact zone for the transfer pool gate is located within storage racks Nos. 1 and 2, which are Region 1 type racks. Cells in rows F through P and 1 through 3 are included (30 cells total).

   2. The primary impact zone for the cask pool gate is located within storage racks Nos. 7 and 8, which are Region II type racks. Cells in rows HH through SS and 51 through 54 are included (44 cells total).

15.1.1.7 Test Equipment Drop

In order to assure that excessive radiological consequences do not occur due to a test equipment skid drop, an analysis was performed that assumes a drop of a 4500 lb. piece of equipment from a height of 47 feet above the pool floor. The assumed equipment consists of a 4-foot by 6-foot base with a 200-inch long vertical H-beam attached to the base at one of the 4-foot edges. Conservative drag calculations made for this piece of equipment indicated that the equipment would impact the top of the racks with a velocity of approximately 206 in/s. The kinetic energy of the equipment is then converted into strain energy in the rack structure.

Calculations were made to determine the load required to compress a fuel rack cell. When the yield point of the cell is reached, local buckling increases rapidly as the cell is compressed. The penetration into the rack top if the equipment base is conservatively assumed to be at an angle with the horizontal of 45° when it impacts the rack was calculated. The maximum penetration in this case is approximately 16 inches.

The top of the Unit 1 fuel assembly is approximately 51.5 inches below the top of the rack. Therefore, this drop will not result in damage to Unit 1 fuel assemblies. The top of the Units 2 and 3 fuel assemblies is approximately 13.2 inches below the top of the rack and this drop would result in damage to 14 Units 2 and 3 fuel assemblies. An additional analysis was made to determine the maximum drop height under which no fuel assembly damage results. It was determined that for a drop height of 72 inches above the top of the rack the test equipment will impact the top of the rack with a velocity of 177 in/s. The penetration into the rack top if the equipment base is conservatively assumed to be at an angle with the horizontal of 45^{o} when it impacts the rack was calculated. The maximum penetration in this case is 13.0 inches below the top of the rack. Given the drop penetration of 13 inches no fuel damage occurs.

Control rods stored integrally with fuel assemblies extend above the top of the fuel assembly upper end fitting about 1.4 inches for a SONGS 1 fuel assembly and 11.1 inches for a SONGS 2 and 3 fuel assembly. Control rods inserted into a SONGS 1 assembly do not increase the potential for fuel damage because a dropped test equipment skid cannot penetrate the racks far enough to impact the top of the control rod. For a SONGS 2 or 3 assembly containing a CEA, analysis shows that the CEA will not be impacted during a potential drop if the test equipment skid is maintained below 11.2 inches above the tops of the racks.

These calculations were done for a Region II rack. Since this type of rack has only one cell wall between adjacent storage locations and the Region I rack has two cell walls between adjacent storage locations, the Region II rack is the limiting case.

Administrative controls will be implemented to provide assurance that the radiological consequences of these drops are acceptable. The administrative controls include the following:

A. The height above the pool floor that the skid may be carried over rack cells which contain Unit 1 fuel assemblies shall be limited to 47 feet (elevation 64 feet 6 inches).

B. When the skid is lowered, it shall be lowered over empty racks or rack cells containing Unit 1 fuel assemblies only.

C. The maximum height that the skid may travel horizontally over the racks containing Unit 2 or 3 fuel assemblies without CEAs shall be 72 inches (elevation 39 feet 10 inches). A drop from this height will not damage Units 2 and 3 fuel assemblies.

D. All Unit 2 or 3 fuel assemblies are to be removed from the test equipment skid impact zone, 10 by 12 cells, prior to lifting or lowering the skid over the high density spent fuel storage racks.

E. The test equipment skid shall be maintained 11 inches or less above the top of the racks when passing over CEA bearing SONGS Units 2 and 3 spent fuel assemblies in the high density spent fuel storage racks.

With these controls in place, it will ensure that the fuel assemblies are not damaged, since the depth of penetration will not impact the racks at the level where the fuel assemblies are located. Since no fuel assemblies are damaged, there are no radiological consequences for the test equipment drop.

15.1.1.8 Spent Fuel Pool Boiling Accident

The postulated loss of all spent fuel pool (SFP) cooling is assumed to result in SFP boiling and the release of a portion of the radionuclide inventory contained in the stored spent fuel assemblies and the SFP water.

The evaluation of the radiological consequences for the SFP boiling event assumes a minimum of 17 months since the shutdown of Units 2 and 3. Appendix 15G identifies the isotopes present in spent fuel after this period of shutdown decay.

Following a loss of SFP cooling, activity releases from the spent fuel due to evaporation and boiling disperse to the Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) locations.

The radiological consequence analysis conservatively does not differentiate between the activity release rates before and after the onset of SFP boiling. Noble gas, iodine and tritium activity present in the assumed fraction of failed fuel rod gap spaces of fuel rods stored within the SFP is released to the SFP water at the noble gas, iodine and tritium escape rate coefficients listed in Chapter 11, with the added conservatism of an assumed spiking factor of 100. The noble gas and iodine fuel rod gap fractions are consistent with Alternative Source Term (AST) methodology. The tritium fuel rod gap fraction is assumed to be the same as that for the majority of noble gas and iodine isotopes. Both before and after the onset of SFP boiling, spent fuel noble gases, iodine and tritium gas escaping from the failed fuel rod gap spaces are assumed to be instantaneously released with no hold up or iodine partitioning in the SFP water.

Tritium activity present in the SFP water prior to the loss of SFP cooling is assumed to be released at the SFP boiling rate for the duration of the event. The SFP boiling rate is conservatively greater than the SFP evaporation rate present prior to the onset of SFP boiling. The SFP boiling rate is a function of the decay heat load, and the heat of vaporization of water.

No credit is taken for activity retention within the fuel handling building air. All activity escaping from the SFP is assumed to be instantaneously released to the environment and atmospherically dispersed to the offsite dose receptors.

Table 15.1-7 lists the parameters used for performing the dose analysis for the postulated SFP boiling event. Additional assumptions associated with Alternative Source Term (AST) modeling are provided in Appendix 15G.

The offsite radiological doses for the postulated SFP boiling accident do not exceed 25% of the 10 CFR Part 50.67 exposure guidelines. The radiological consequences of this event are presented in Table 15.1-8.

Table 15.1-7
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SPENT FUEL POOL BOILING EVENT

SFP BOILING PARAMETER	MODELED VALUE
Number of Stored Spent Fuel Assemblies in Spent Fuel Pool (SFP)	1542 (= total available spaces)
SFP Decay Heat Load (17 months post-shutdown), BTU/hr	3.879E6
SFP Boiling Rate, ft^3/hour	66.84
Failed Fuel, %	1
Failed Fuel Escape Rate Coefficients, sec^{-1} Noble Gases Iodine Tritium	 6.5E-8 1.3E-8 1.4E-11
Spiking Factor for Noble Gases, Iodine and Tritium Releases from Failed Fuel	100
Fuel Rod Gap Release Fractions Iodine-131 Krypton-85 Other noble gases (Krypton, Xenon) Other Halogens (Iodine, Bromine) Tritium	 0.08 0.10 0.05 0.05 0.05
SFP Water Iodine Decontamination Factor	1
Activity Release from FHB	SFP Activity Release instantaneously dispersed to dose receptor
Offsite Dose Evaluation Model	per Appendix 15G

Table 15.1-8
RADIOLOGICAL CONSEQUENCES OF SPENT FUEL POOL BOILING

DOSE RECEPTOR	SFP BOILING DOSE (REM TEDE)	ACCEPTANCE CRITERION (REM TEDE)
EAB (Maximum 2-hour dose -- 0.0 to 2.0 hours)	0.08E-03 (0.08 mRem TEDE)	6.3
LPZ (30-day accident duration)	0.25E-03 (0.25 mRem TEDE)	6.3

15.1.1.9 Spent Fuel Assembly Drop

15.1.1.9.1 Spent Fuel Assembly Drop onto Reconstitution Station

15.1.1.9.1.1 Introduction

As discussed in Chapter 9, a spent fuel assembly will be placed atop a rack spacer during fuel reconstitution. As a result, the spacer will raise the top of the spent fuel assembly above the top of the high density spent fuel storage racks. In the reconstitution station when the rack spacers are used, there is a greater potential for damage to spent fuel assemblies than in locations where the spacers are not used, if a spent fuel assembly is accidentally dropped.

15.1.1.9.1.2 Summary of Methods

Current procedures restrict the number of spent fuel assemblies in the reconstitution station to six. A situation could exist during fuel reconstitution where five spent fuel assemblies are placed atop rack spacers and the sixth one is being moved towards the reconstitution station and is therefore above one or all of these elevated spent fuel assemblies. The drop of a spent fuel assembly onto the reconstitution station could damage the dropped assembly, as well as some of the spent fuel rods in the assemblies located on spacers within the reconstitution station.

To prevent such an accident in the reconstitution station during fuel reconstitution, the following two administrative controls are implemented:

A. No spent fuel assembly shall be moved over any spent fuel assembly in the reconstitution station or over adjacent storage locations when spent fuel assemblies are in the reconstitution station on the rack spacers.

B. No CEA bearing spent fuel assemblies shall be placed atop rack spacers in the reconstitution station.

Additionally, a fuel assembly dropped onto the spent fuel storage racks could have the potential to topple over onto a fuel assembly located on a spacer in the reconstitution station. In this event the fuel assembly in the reconstitution station is not damaged. The damage to the dropped fuel assembly is addressed in Section 15.1.1.4.

15.1.1.9.1.3 Results

Analyses were completed which evaluated the potential for damage to spent fuel assemblies located on reconstitution spacers in a reconstitution station. Results concluded that damage to spent fuel could occur if a spent fuel assembly is moved and subsequently dropped onto fuel located in the reconstitution station.

15.1.1.9.1.4 Conclusion

The administrative controls stated above will preclude damage to spent fuel assemblies during reconstitution.

15.1.1.9.2 Spent Fuel Assembly Drop onto CEA Bearing Spent Fuel Assemblies

Control Element Assemblies (CEAs) which have been replaced are stored in the spent fuel pool inserted into spent fuel assemblies. The top of a CEA comes within 2.11 inches of the top of the high density spent fuel storage rack when stored integrally with a SONGS Units 2 and 3 spent fuel assembly. The maximum potential damage occurring in the event of a spent fuel assembly drop onto a CEA bearing spent fuel assembly, would be failure of all fuel rods in the dropped fuel assembly and all fuel rods in the impacted CEA bearing spent fuel assembly.

The radiological consequences for the failure of two fuel assemblies is addressed by the postulated fuel handling accident inside the fuel handling building in UFSAR Section 15.1.1.4.

15.1.1.10 Use of Miscellaneous Equipment Under 2000 lbs

15.1.1.10.1 Introduction

Several miscellaneous pieces of equipment weighing less than 2000 lbs (e.g., equipment used for ultrasonic testing, gamma spectrometer, eddy current testing, periscope, oxide measurement devices, tools and work platforms used for fuel reconstitution, temporary underwater pumps, skimmers) are required to be located or moved over the spent fuel storage racks during fuel handling operations and normal spent fuel pool maintenance. Administrative controls are currently placed on the movement of loads in excess of the nominal weight of a fuel assembly, CEA, and associated handling tool over other fuel assemblies in the storage pool. Hence, this evaluation is based on the potential for damaging fuel assemblies, if it is postulated that

equipment which does not exceed the weight of a fuel assembly, CEA, and associated handling equipment (i.e., less than 2000 lbs) is dropped onto other spent fuel assemblies.

15.1.1.10.2 Summary of Methods

The restrictions on movement of loads in excess of the nominal weight of a fuel assembly, CEA, and associated handling tool over other fuel assemblies in the storage pool ensure that, if this load is dropped, this event is bounded by other load drop events. Specifically, the activity release would be less than a fuel handling accident inside the fuel handling building, which analyzes a drop weight of 2065 pounds, which represents a bundle dry weight of 1495 pounds, plus 120 pounds of components (e.g., control element assembly [CEA], neutron sources, etc.) plus 400 pounds of grapples, plus 50 pounds discretionary margin. Moreover, any possible distortion of fuel contained in the racks would not result in a critical array.

The restrictions on movement of loads in excess of 2000 pounds, the nominal weight of a fuel assembly, CEA, and associated handling tool are administratively controlled.

Since this event is bounded by another more limiting event, there are no principal assumptions, inputs, or sequence of events to present.

15.1.1.10.3 Results

The administrative controls relative to moving loads less than 2000 lbs minimize the potential for radiological release or criticality events. However, the dose consequences of load drops on spent fuel contained in storage pool racks are bounded by the postulated fuel handling accident inside the fuel handling building in UFSAR Section 15.1.1.4.

15.1.1.10.4 Conclusion

The dose consequences of load drops on spent fuel contained in storage pool racks are bounded by the postulated fuel handling accident inside the fuel handling building in UFSAR Section 15.1.1.4.

REFERENCES

1. Gabrielson, V. K., SHOCK - A Computer Code for Solving Lumped Mass Dynamic Systems, SCL-DR-65-35, January 1966.