San Onofre Nuclear Generating Station
Unit 2 & 3
Updated Final Safety Analysis Report
Revised April 2011

<u>Chapter 5.0 - Reactor Coolant System and Connected Systems</u>

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function as required by the specifications. The vibration levels are monitored during this test. Evidence of the pumps operating near a critical speed would be noted as excessive vibration.

Full scale seal testing is performed at rated pressure, temperature, water chemistry, and speed to demonstrate the capability of the seals to satisfactorily perform their design function.

5.4.2 STEAM GENERATORS

5.4.2.1 Design Bases

The two steam generators are designed to transfer 3458 MWt from the RCS to the secondary system, producing approximately 15.176 x 10^6 lb/h of 833 lb/in.^{2}a saturated steam, when provided with 442°F feedwater. The saturated steam pressure of 833 psia is the best estimate pressure at the steam generator outlet nozzle with the reactor coolant inlet temperature at 598 °F, reactor coolant best estimate flow rate, 0% tubes plugged and an assumed tube fouling factor. The actual steam outlet pressure will vary depending on the actual values of these parameters during plant operation. Moisture separators and steam driers in the shell side of the steam generator limit the moisture content of the steam to 0.10 wt% during normal operation at full power. The steam generator design parameters are listed in table 5.4-4. The steam generators, including the tubes, are designed for the RCS transients listed in paragraph 3.9.1.1 so that the code allowable stress limits are not exceeded for the specified number of cycles. All transients have been established based on conservative assumptions of operating conditions in consideration of supportive system design capabilities. The steam generators are capable of sustaining the following additional design transients without exceeding code allowable stress limits:

A. Ten primary side hydrostatic tests with the primary side pressurized to 1.25 times the design pressure and the secondary side at atmospheric pressure.

B. Ten secondary side hydrostatic tests with the secondary side pressurized to 1.25 times the design pressure and the primary side at atmospheric pressure.

C. Two hundred primary side leak tests with the primary side at the operating pressure of 2250 lb/in.^{2}a and the secondary side at atmospheric pressure.

D. Two hundred secondary side leak tests with the secondary side at 900 lb/in.^{2}a and the primary side at atmospheric pressure.

E. Fifteen thousand cycles of adding 40°F feedwater at 820 gal/min to the steam generators through the main feedwater nozzle when at hot standby conditions (normal condition). The basis is nominal operating conditions assuming intermittent feeding of the steam generators.

F. Eight cycles of adding 40°F feedwater at 700 gal/min to the steam generator after a loss of normal feedwater. This is based on the quantity of water needed to bring the plant to shutdown cooling initiation temperature with intermittent feeding between the low water level and normal water level.

G. Four thousand pressure transients of 85 $lb/in.^2$ across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps (normal condition).

Table 5.4-4

STEAM GENERATOR PARAMETERS[(a)]

Parameter	Value
Number of SGs per plant unit	2
Heat transfer rate, each, Btu/h	5.900×10^9
Primary side	
Design pressure/temperature, lb/in.^{2}a/°F	2500/650
Coolant inlet temperature, °F	598.0
Coolant outlet temperature, °F	541.3
Coolant flow rate, each, lb/h	79.79×10^6 [(b)]
Coolant volume at 68°F each, ft^3	2003
Tube size, OD, in.	0.75
Tube thickness, nominal, in.	0.0429
Secondary side	
Design pressure/temperature, lb/in.^{2}a/°F	1100/560
Steam pressure at steam nozzle outlet lb/in.^{2}a	833[(b)]
Steam flowrate (with 0.10% moisture), lb/h	7.588×10^6 [(b)]
Feedwater temperature at full power, °F	442[(b)]
Moisture carryover, by weight (maximum), %	0.10
Primary inlet nozzle, No./ID, in.	1/42
Primary outlet nozzle, No./ID, in.	2/30
Steam nozzle, No./ID, in.	1/38
Feedwater nozzles, No./NPS/schedule	1/18/100
Overall heat transfer coefficient (estimated), Btu/hr-ft^2-°F	1280
Normal Blowdown flow, lb/hr	0.155×10^6

(a) The steam generators are qualified to operate in the T_{hot} range from 598 to 611°F, which corresponds to the T_{cold} range from 541.3 to 555.4 °F.

(b) The values of these parameters represent the best estimate values for a single steam generator based on full power operation with the reactor coolant inlet temperature at 598°F, 0% tubes plugged and an assumed tube fouling factor. The actual values of these parameters may be different than these listed in the table, depending on the steam generator condition.

The principal ferritic materials for the fabrication of the primary coolant boundary in the steam generator were specified to ASME Code, Sections II and III, 1998 Edition through 2000 Addenda requirements. The fracture toughness results met these requirements. The specific materials for the steam generator tubes are provided in subsection 5.2.3.

The operating pressure and temperature limits for the steam generator primary side were determined in accordance with 10CFR50 Appendix G, and the ASME Code, Section III, Appendix C.

The materials used on the secondary side of the steam generator were ordered to ASME Code, Sections II and III, 1998 Edition through 2000 Addenda. The operating pressure and temperature limits for the steam generator secondary side are addressed in Section 10.3.6.1.

The method of fastening tubes to the tube sheet conform with the requirements of ASME Code, Sections III and IX. Tube expansion into the tube sheet is total with no voids or crevices occurring along the length of the tube in the tube sheet. Tube supports are of the plate type with broached trefoil flat-land tube holes which provides no crevice or low flow areas that might promote accumulation of the corrosion products.

The steam generator was designed to ensure that critical vibration frequencies are well out of the range expected during normal operation and during abnormal conditions. The tubing and tubing supports are designed and fabricated with considerations given to both secondary side flow induced vibration and reactor coolant pump induced vibrations. In addition, the steam generator assemblies are designed to withstand the blowdown forces resulting from the severance of the steam nozzle. The steam generator assemblies are also designed to withstand the severance of any one of the feedwater nozzles. The two accidents are not considered simultaneously.

Discussion of the techniques used to maintain cleanliness during final assembly and shipment are discussed in subsection 5.2.3 and appendix 3A.

Onsite cleaning and cleanliness control procedures for the steam generator are consistent with the recommendations of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants and ANSI N45.2-1973, Cleaning of Fluid Systems and Associated Components For Nuclear Power Plants.

5.4.2.2 Description

The steam generator is a recirculating, vertical U-tube type heat exchanger converting feedwater into saturated steam. The steam generator vessel pressure boundary is comprised of the channel head, lower shell, middle shell, transition cone, upper shell and upper head. The steam generator internals include the divider plate, tubesheet, tube bundle, feedwater distribution system, moisture separators, steam dryers and integral steam flow limiter installed in the steam nozzle. The channel head is equipped with one reactor coolant inlet nozzle and two outlet nozzles. The upper vessel is equipped with the feedwater nozzle, steam nozzle and blowdown nozzle. In the channel head, there are two 18 inch access manways. In the upper shell, there are two 16 inch access manways. The steam generator is equipped with six (6) handholes and 12 inspection ports providing access for inspection and maintenance. In addition, the steam generators are equipped with several instrumentation and minor nozzles for layup and chemical recirculation intended for chemical cleaning. The steam generator is illustrated in figure 5.4-6.

Reactor coolant enters the channel head at the bottom of each steam generator through the single inlet nozzle, flows through the U-tubes, and leaves through the two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums in the channel head.

Feedwater enters the steam generator through the feedwater nozzle where it is distributed via a feedwater distribution ring. The feedwater ring is made of erosion-corrosion resistant pipe and fittings and is designed to minimize the potential for waterhammer and thermal stratification. The feedwater is distributed at low velocity from the perforated spray nozzles mounted on the top of the feedwater ring. This design prevents foreign objects from entering the tube bundle and feedwater from impinging the surrounding components.

The feedwater leaving the spray nozzles mixes with the recirculated water and enters the downcomer, which is an annular space between the steam generator shell and a wrapper enclosing the tube bundle. At the bottom of the downcomer, the water changes the direction and enters the evaporator area, and flowing upwards picks up heat from the reactor coolant flowing in the U-tubes. The water/steam mixture exits the evaporator and enters into 38 centrifugal moisture separators. Upon leaving the separators, high quality steam enters a set of eight (8) banks of single-tier, chevron type steam dryers where its moisture content is lowered to less than 0.10% by weight at full power steam flow rate.

Impurities from the secondary side fluid are being removed by a means of continuous blowdown. The blowdown nozzle is located in the tubesheet and serves also as a secondary side drain nozzle.

The steam generator supports are described in subsection 5.4.14. Secondary side overpressure protection is provided by 18 spring-loaded ASME Code safety valves mounted on the main steam lines as described in subsection 5.4.13.

5.4.2.3 Evaluation

5.4.2.3.1 Steam Generator Tubes

5.4.2.3.1.1 Chemistry Compatibility

The steam generator, tubes are 0.75-inch OD with the wall thickness of 0.0429 wall thickness and are made of thermally treated Alloy 690. Tube sizing incorporates a general corrosion allowance that will provide for operation over the plant design lifetime. The steam generator tube support plates are made of Type 405 stainless steel. The combination of these materials and the tube support design is intended to minimize the potential for tube denting due to deposition of the general corrosion products on the tube support plates. In the San Onofre steam generators, feedwater chemistry control is an all-volatile chemistry control, which is discussed in Subsection 10.3.5

A discussion of chemistry control and corrosion control effectiveness to preclude denting is provided in subsections 10.3.5 and 10.4.5. In addition, corrosion is further inhibited by the use of condenser tubes that are made of titanium as discussed in paragraph 10.4.1.2.1.

5.4.2.3.1.2 Mechanical Considerations

The reactor coolant pumps have a rotational speed of 1180 r/min, (with 1.0 specific gravity water), therefore imposition of exciting frequencies of 19 to 20 Hz and 95 to 100 Hz was considered in the steam generator design.

The low frequency range is defined as a mechanical vibration resulting from the transmission of a mechanical impulse at the frequency of pump rotation. The upper frequency range is defined as a sinusoidal pressure vibration of ± 6 lb/in.^{2} in the reactor coolant piping that contains the pump. The pressure variation results from the impeller vanes interacting with the cut-water vane at the volume outlet during each revolution of the impeller.

5.4.2.3.1.3 Tube Wall Thinning

The extent of tube wall thinning that can be tolerated in the San Onofre steam generators without exceeding the allowable stress limits is determined by structural analysis in accordance with the requirements of the ASME Section III and USNRC R.G. 1.121. The analytical model for this analysis includes the tubes and tube supports only. The scope does not include the steam generator upper internals (moisture separators or steam dryers), because those components are not connected to the tube bundle and loads cannot be transferred from these components to the tube bundle.

The structural analysis is performed in two parts. The first part is a Functional Integrity Evaluation of a non-degraded tube with the nominal wall thickness when subjected to an upper bound plant Faulted condition loading. The resulting stresses are calculated and compared to the applicable Code allowable limits. This part of the analysis demonstrates that the non-degraded tube has a significant structural margin.

The second part of the analysis is a Degraded Tube Evaluation, in which the minimum tube wall thickness required to meet the structural requirements of R.G.1.121 is calculated. This evaluation considers: (1) wall thickness loss over the entire tube length, (2) wall thickness loss at the tube intersections with tube support plates (TSPs), and (3) wall thickness loss at the tube intersections with the anti-vibration bars (AVBs) in the tube bundle U-bend region. The minimum wall thickness is calculated for: (1) the Faulted condition, and (2) the Normal operating condition. The more limiting of these two loading conditions determines the minimum allowable tube wall thickness. As a check to verify the acceptability of the calculated minimum wall thickness, a structural analysis is performed to demonstrate that the degraded tube with the minimum wall thickness will not burst or collapse under the conditions specified in R.G. 1.121.

The following paragraphs provide a summary of the methods of evaluation used in both parts of the structural analysis.

A. Functional Integrity Evaluation

This part of the analysis evaluates overall tube integrity to show that the primary stresses in the non-degraded tube with the nominal wall thickness are within the Code allowable limits. The tube primary membrane and bending stresses are evaluated under the Faulted condition, which considers a limiting combination of the design basis events – loss of coolant accident (LOCA), design bases earthquake (DBE) and steam line break (SLB). In this case, the SLB is conservatively represented by the pressure differential resulting from the primary side being at the design pressure and the secondary side being at atmospheric pressure.

The CEFLASH computer code was originally used to perform the hydraulic dynamic analysis of the primary coolant loop during a LOCA event with the original steam generators installed in order to determine dynamic structural response of the steam generator to the impulsive loading imparted by the escaping fluid. This loading was calculated at various original steam generator elevations, assuming a double-ended guillotine break in the reactor coolant cold leg piping.

For the replacement steam generators, a structural model of the tube bundle consisting of six groups of tubes in the upper part of the bundle (the top two tube support spans plus the U-bend) and two straight tube assemblies in the lower part of the bundle is generated using the ANSYS computer code. Using this model, the seismic and LOCA rarefaction analysis is performed. For the seismic analysis, the seismic response spectra developed for the replacement steam generators are used. For the rarefaction wave analysis, the pressure-time history (dynamic loadings) previously developed for the original steam generators (as described in the paragraph above) is used. For stress calculation, the maximum seismic and LOCA rarefaction wave loads are combined using the square-root-of-the-sum-of-the-squares (SRSS) method.

In this analysis, the SLB event is conservatively modeled by assuming the primary side being at the design pressure, plus the pressure relief valve accumulation, and the secondary side being at atmospheric pressure, and applying this pressure differential as a step function. The pressure stresses calculated based on this differential pressure are added directly to the combined stresses due to the seismic and LOCA loads.

The resulting tube primary stresses are then compared to the Code allowable stress limits for the Faulted Condition.

B. Degraded Tube Evaluation

The degraded tube evaluation is performed in accordance with the requirements of R.G. 1.121 by first calculating the minimum tube wall thickness required to meet the Code allowable stress limits for the limiting plant condition. The limiting plant condition is the Normal operating condition (due to the lowest allowable) and this condition is the basis for establishing the minimum degraded tube wall thickness.

The calculations are performed in accordance with the R.G.1.121 requirement that the margin of safety against tube rupture under normal operating conditions be no less than 3 at any tube location where defects have been detected.

For any tube size and material, the minimum wall thickness required to meet the Code allowable stress limits depends on the pressure differential across the tube wall. For the purpose of this analysis, the highest expected primary-to-secondary pressure differential under the Normal operating condition is used for conservatism. This pressure differential is based on the lowest steam generator secondary pressure, which is expected to occur at the steam generator end-of-life.

The minimum wall thickness calculations consider a case where the tube wall is uniformly thinned along its entire length. This case is limiting and is the basis for establishing the minimum tube wall thickness. In addition, two other cases are evaluated for future reference. These cases are for limited axial length degradation at two distinct locations where tube thinning is most likely to occur – at the TSP intersections and at the AVB intersections. According to NUREG/CR-0718, tubes with shorter degradation lengths have higher burst pressures. The length-to-burst pressure relation from this reference is used to calculate the minimum wall thickness for these additional cases.

The degraded tube (having the minimum wall thickness calculated as described above) is evaluated against the allowable stress limits for primary membrane plus bending (in-plane) stress intensity in the straight leg region (including TSP intersections) and in the U-bend region (at AVB intersections). The stresses at the degradation locations are calculated by multiplying the stresses for the non-degraded tube by the ratio of the corresponding section properties of the nominal and degraded tube.

The resulting tube primary stresses are then compared to the Code allowable stress limits for the Faulted condition to demonstrate that the degraded tube will not burst under this worst case loading.

The degraded tube is also evaluated against collapse under the maximum possible secondary-to-primary pressure differential during a LOCA. For this purpose, calculations are performed for an ovalized tube, as the collapse pressure for such tubes is lower than for the perfectly round tubes.

C. Summary of Results

The results of the functional integrity evaluation indicate that for the non-degraded tube with a nominal wall thickness of 0.0429 inch under the hypothetical combined loading resulting from LOCA+DBE+SLB, the maximum stress intensity occurs at the uppermost TSP and is 35.1 ksi. This stress intensity compares favorably with the Code primary membrane plus bending stress allowable of 75.4 ksi for the Faulted condition. The primary-to-secondary pressure differential used to model the SLB event was conservatively taken as 2560 psid, which is the primary side at the design pressure of 2500 psia plus the 3% primary side relief valve accumulation, and the secondary side at atmospheric pressure.

The results of the degraded tube evaluation indicate that the minimum allowable tube wall thickness is 0.01923 inch for uniform degradation along the entire tube length. This result applies to any tube bundle region, straight leg or U-bend, where degradation axial length is 1.5 inch, or greater. For the TSP intersection, where the degradation length is assumed to be equal to the thickness of the TSP (1.38 inch), the minimum wall thickness is 0.01895 inch. For the AVB intersection, where the degradation length is assumed to be equal to the average tube-to-AVB contact length, the minimum wall thickness is 0.01526 inch. Therefore, the degraded tube minimum wall thickness is conservatively taken as 0.01923 inch, which corresponds to 55.17% tube wall thinning.

The stress evaluation indicates that the degraded tube with a minimum wall thickness of 0.01923 inch, under the hypothetical combined loading resulting from LOCA+DBE+SLB (2560 psid), will have a maximum stress intensity of 46.5 ksi in the straight leg (at the uppermost TSP) and 45.6 ksi at the limiting U-bend location. These stress intensities compare favorably with the Code allowable limits for the Faulted condition of 73.2 and 74.3 ksi, respectively.

The ASME Code criterion for tube collapse is that maximum pressure differential across the tube wall be no greater than 90% of the pressure gradient that the degraded tube has to withstand without collapsing. The maximum calculated secondary-to-primary differential pressure during a LOCA for San Onofre steam generators is 931 psid. This means that the minimum differential pressure that the degraded tube is required to withstand without collapsing is 1035 psid. Based on the test data, the differential pressure required to collapse a uniformly degraded tube with the minimum wall thickness of 0.01923 in. and the ovality of 2.8% (the maximum ovality specified for the replacement steam generators) is 1210 psid. This compares favorably with the above Code criterion.

In conclusion, a degraded tube with a minimum wall thickness of 0.01923 inch over its entire length meets the ASME Code general primary membrane stress criteria, which ensure that such a tube satisfies the requirements of R.G.1.121.

In addition to the above analyses, the following criteria from Reference 3 must be met in determining the allowable tube thinning:

1. Tubes with detected acceptable defects will not be stressed during the full range of normal reactor operation beyond the elastic range of tube material.

2. Crack-type defects that could lead to tube rupture either during normal operation or under postulated accident conditions are not acceptable.

When evaluating against the above criteria for San Onofre steam generator geometry and operating conditions, tube thinning of 55% is allowable for all tubes.

In establishing the Technical Specification limits for Units 2 and 3 tube inspections, the NRC imposed an across the board 20% reduction in the allowable tube thinning. Half of this reduction accounts for tube continuous degradation growth during the operational cycle prior to the next inspection. The other half of this reduction accounts for the accuracy of the wall thinning measurement technique. Based on the above, the Technical Specification tube plugging limit for San Onofre Units 2 and 3 is set at 35%.

5.4.2.3.2 Potential Effects of Tube Rupture

The steam generator tube rupture incident is a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube would allow for the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with water in the shell side of the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser or directly to the condenser via the steam dump bypass system. Non-condensable radioactive gases in the condenser would be removed by the main condenser evacuation system and discharged to the plant vent stack. Analysis of a steam generator tube rupture incident, assuming complete severance of a tube, is presented in section 15.6.

Experience with the nuclear steam generators indicates that the probability of complete severance of a tube is remote. The strength of the material used to fabricate the steam generator tubes ensures that a double-ended tube rupture is extremely unlikely. The more probable modes of failure, which result in smaller penetrations, are those involving the occurrence of wear pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet. Detection and control of steam generator tube leakage is described in subsection 5.2.5.

5.4.2.3.3 Composition of Secondary Fluid

The concentration of radioactivity in the secondary side of the steam generators is dependent upon the concentration of radionuclides in the reactor coolant, the primary-to-secondary leak rate, and the rate of steam generator blowdown. The expected specific activities in the secondary side of the steam generators during the periods of normal operation are given in table 11.2-27.

Activities are based on operations with average defective fuel cladding, a total primary-to-secondary leakage of 100 lb/day, and 60 gal/min blowdown rate to the blowdown processing system. An evaluation of the shell side radioactivity is presented in section 11.2. Limits for radioactivity levels in the secondary side of the steam generators and the bases for these limits are provided in the Technical Specifications.

The recirculation water within the steam generators will contain volatile additives necessary for proper chemistry control. These and other chemistry considerations of the main steam system are discussed in subsection 10.3.5.

5.4.2.3.4 Tube Support Plate Thinning

The tube supports are made of Type 405 stainless steel which eliminates the possibility of support plate thinning due to general corrosion.

5.4.2.3.5 Tube Repair

The periodic inspections of steam generator tubes result in certain tubes that exceed established criteria for remaining in service. These tubes will be removed from service by plugging.

5.4.2.3.6 Nozzle Dams

Access for tube inspection and repair during a refueling outage is provided by installing seals, known as steam generator nozzle dams, in the hot leg and cold leg nozzles. With the primary head isolated from the rest of the RCS, the inside of the steam generator may be kept dry while the RCS water level is returned to the refueling level.

The nozzle dam system is comprised of an elastomeric diaphragm supported by interlocked aluminum dam segments, which are installed in the mounting grooves integral to the channel head nozzles. Leakage of RCS water is prevented by two redundant inflatable seals and one passive emergency seal. RCS venting configurations are controlled to ensure that the effective vent area is sufficient to prevent overpressurization of dams. The aluminum dam segments and latching mechanism are designed to withstand the highest RCS pressure that might be generated in the unlikely event that shutdown cooling were lost while the dams are installed.

5.4.2.4 <u>Tests and Inspections</u>

5.4.2.4.1 Fabrication Tests and Inspections

The steam generator is tested in accordance with ASME Boiler and Pressure Vessel Code, Section III. The following nondestructive tests, some of which were not required by the code, were performed during fabrication.

Components	Test[a]
Tube-sheet forging	UT, MT, PT
Tubesheet Cladding	UT, PT
Channel Head Forging	UT, MT
Channel Head Cladding	UT, PT
Secondary Shell and Head Forging	UT, MT
Tubes	UT, ET
Nozzles End (Forging)	UT, MT
Studs	UT, MT
Welds	
Shell, circumferential	RT, MT, UT
Cladding	UT, PT
Nozzles to shell	MT, UT, PT
Tube-to-tube sheet	PT
Instrument connections	MT
All welds - after hydrostatic test	MT
Nozzle safe ends	RT, (MT or UT)
Level Taps	MT

[a] UT = Ultrasonic testing
MT= Magnetic-Particle testing
RT = Radiographic testing
PT = Liquid-penetrant testing
ET = Eddy-current testing

During design and fabrication of the steam generator, additional operations beyond the requirements of the ASME Boiler and Pressure Vessel Code, Section III, were performed by the vendor. These included ultrasonic testing for defects in tube sheet clad and ultrasonic testing of weld clad for bond integrity.

Initial hydrostatic tests of the primary and secondary sides of the steam generator are conducted in accordance with ASME Code, Section III. Leak tests are also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Steam generator performance is further verified during the initial startup tests. Provisions for onsite cleaning and cleanliness control are described in subsection 5.2.3.

5.4.2.4.2 Steam Generator Inservice Inspection

5.4.2.4.2.1 Preservice Examination

As stated in subsection 5.2.4, the preservice examination for Class 1 steam generator components complies with the requirements of ASME Section XI, 1998 Edition through 2000 Addenda.

The preservice and inservice inspection programs for examining steam generator tubes were consistent with the recommendation of Regulatory Guide 1.83, Revision 1. The preservice examination requirements for steam generator tube inspection have been adequately met and all unacceptable defects were eliminated in accordance with ASME Section III.

5.4.2.4.2.2 Inservice Inspection

The ASME Section XI Code inservice inspection (ISI) requirements for the Class 1 primary side of the Steam Generators in Units 2 and 3 are defined in subsection 5.2.4. The specific examination and pressure test requirements are defined in the ISI Program Plans for the inspection intervals.

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 Design Basis

The reactor coolant loop piping is designed and analyzed for normal operation and all transients discussed in subsection 3.9.1 and the following additional requirement. During heatup and cooldown of the plant, the allowable rate of temperature change for the surge line is increased to 200°F/h as a design requirement specified in paragraph 3.9.1.1. Loading combinations and stress criteria associated with faulted conditions are presented in paragraph 3.9.3.1. In addition, certain nozzles are subjected to local transients that are included in the design and analysis of the areas affected. Thermal sleeves were installed in the surge nozzle, safety injection nozzles, and charging nozzle to accommodate these additional transients. Surge line and safety injection thermal sleeves are not required for the nozzle to meet the design stress requirements and some are no longer installed. Principal parameters are listed in table 5.4-5. The ASME code and addenda the piping is designed to is specified in subsection 5.2.1.

In addition to being specified as Seismic Category I, the following additional vibratory piping assemblies are designed so that no damage to the equipment is caused by the frequency ranges of 19 to 20 Hz and 95 to 100 Hz. The definitions of these frequencies are the same as for the steam generator. Additional presentation relating to seismic and dynamic analysis and criteria for the reactor coolant piping is contained in subsections 3.7.2 and 3.9.2, respectively.

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Revised April 2011

Chapter 15.0 – Accident Analyses

Section 15.6.3.2 – Steam Generator Tube Rupture

15.6.3.2 Steam Generator Tube Rupture

15.6.3.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a steam generator tube rupture with or without a concurrent loss of normal AC power (LOAC) classifies it as a limiting fault incident as defined in reference 1 of section 15.0. The worst case of an SGTR with LOAC is presented below. The steam generator tube rupture accident is a penetration of the barrier between the RCS and the main steam system and results from a failure of a steam generator U-tube. In terms of break size, the worst case is a postulated double-ended tube rupture. Experience with nuclear steam generators indicates that the probability of complete severance of the Inconel vertical U-tubes is remote. No such double-ended rupture has ever occurred in a steam generator of this design. The more probable modes of failure result in considerably smaller penetrations of the pressure barrier. They involve the formation of etch pits or small cracks in the U-tubes or cracks in the welds joining the tubes to the tube sheet.

In accordance with the direction given in Sections 15.0 & 15.0.7, additional information which completes the presentation of this event is provided in Section 15.10.6.3.2.

15.6.3.2.2 Sequence of Events and Systems Operations

Integrity of the barrier between the RCS and main steam system is significant from a radiological standpoint since a leaking steam generator tube would allow transport of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with shellside water in the affected steam generator. During normal plant operations, some of this radioactivity would be transported through the turbine to the condenser where the noncondensible radioactive materials would be released via the condenser air ejectors.

Since the plant is operating at 100% power for approximately 16 minutes before the effects of the primary-to-secondary leak cause the reactor trip, the radioactivity concentration in the steam generators is allowed to increase before the steam generator safety valves open releasing radioactive materials to the atmosphere.

Following the tube rupture, the RCS pressure would gradually decrease. The primary-to-secondary leak rate and drop in RCS pressure would result in all CVCS charging pumps being brought on line and reactor trip due to low pressurizer pressure. Following reactor trip, the main steam system pressure would increase to the point where the turbine bypass valves would open to control the main steam system pressure. If turbine bypass is unavailable, the steam generator safety valves would open to control the main steam system pressure. The operator can isolate the damaged steam generator and cool the NSSS using manual operation of the auxiliary feedwater and the atmospheric steam dump valve of the unaffected steam generator any time after reactor trip occurs. The analysis presented herein conservatively assumes that operator action is delayed until 30 minutes after first indication of the event.

Diagnosis of this accident would be facilitated by radiation monitors in the blowdown sample lines from each steam generator, in the blowdown processing system neutralization sump discharge sea line which processes blowdown from both steam generators, in the condenser air ejector discharge line, and adjacent to the main steam lines. These monitors would initiate alarms in the control room and inform the operator of abnormal activity levels and that corrective action is required.

Behavior of the systems varies depending upon the size of the rupture. For leak rates up to the capacity of the charging pumps in the CVCS, reactor coolant inventory can be maintained and an automatic reactor trip would not occur. During the first 30 minutes of the accident, a reactor trip is not necessary because the safety limits are not approached and there is no danger of violating dose limits. The 30-minute interval is a conservative time period based on the availability of alarms and indications. The Technical Specifications specify plant shutdown within 40 hours of detection for leaks greater than 1 gal/min. In addition, the plant emergency procedures will provide for rapid plant shutdown based on operator action in the event of a tube failure. Assuming a 30-minute operator action interval, the operator can then take action to ramp down the plant or to manually trip the reactor and place the plant in cold shutdown. Under these operating conditions, the gaseous fission products would be released from the main steam system at the condenser air ejector discharge until the shutdown cooling system is initiated.

For leaks that exceed the capacity of the charging pumps, pressurizer water level and pressurizer pressure decrease and a reactor trip results.

Table 15.6-5 gives a sequence of events which occur following a steam generator tube rupture with concurrent loss of normal AC power.

15.6.3.2.3 Core and System Performance

A. Mathematical Model

The NSSS response to a steam generator tube rupture with concurrent loss of normal AC power was simulated using the CESEC computer program described in section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with the CE-1 CHF correlation described in chapter 4.

B. Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a steam generator tube rupture with concurrent loss of normal AC power are discussed in section 15.0. In particular, those parameters which were unique to the analysis discussed below are listed in table 15.6-6.

The initial conditions for the principal variables monitored by the core operating limit supervisory system (COLSS) were varied over the reactor operating space given in table 15.0-5 to determine the set of conditions which would produce the most adverse consequences following a steam generator tube rupture with a concurrent loss of normal AC power. Various combinations of initial core inlet temperature, core inlet flowrate, and pressurizer pressure were considered. In addition, the scram reactivity used was consistent with the axial power distribution used. Decreasing the initial core inlet temperature increases the primary-to-secondary leak rate and integrated leak, but reduces the releases via the steam generator safety valves. Since the steam generator pressure and temperature are initialized at lower values, a point is reached where the steam generator can increase and peak without opening the steam generator safety valves. Decreasing the RCS pressure hastens the low pressurizer pressure reactor trip, but results in lower releases due to a lower leak rate. Therefore the initial RCS pressure assumed, which is in fact outside the space given in table 15.0-5, ensures that the results are conservative. Increasing the core inlet flowrate produces faster energy transport through the RCS and results in an increased leak rate and higher releases from the steam generator safety valves. Varying the primary-to-secondary break size can produce a case showing higher offsite dose for a break size smaller than that equivalent to a double-ended rupture. For break sizes resulting in a reactor trip during the first 30 minutes of the incident (see emergency procedures), the initial leak rate decreases from that value equivalent to a double-ended rupture, and the offsite dose also decreases due to the drop in the integrated leak. The decrease in break size also delays the time of reactor trip. As the break size is decreased further, the integral leak is reduced for the 30-minute operator action interval, and, therefore, the radiological consequences will be less severe. For the smaller breaks sizes, the following information is still available to the operator:

1. Radiation monitors

2. Difference in steam generator water levels or feedwater flowrates if the automatic steam generator control is being used

3. All CVCS charging pumps on

4. Drop in RCS pressure

Table 15.6-5

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE WITH CONCURRENT LOSS OF NORMAL AC POWER

Time (seconds)	Event	Setpoint or Value
0.0	Tube rupture occurs	----
520.6	Pressurizer heaters de-energized, ft^3(8.9 ft below 100% power operating level)	334
985.1	Low pressurizer pressure boundary trip signal generated by the CPC, turbine stop valves close, loss of normal AC power, psia	1,785
985.5	CEAs begin to drop into core	----
991.3	No. 2 steam generator safety valves begin to open, psia	1,089
991.6	No. 1 steam generator safety valves begin to open, psia	1,089
995.3	Maximum No. 1 steam generator pressure, psia	1,131
995.3	Maximum No. 2 steam generator pressure, psia	1,131
1,000.0	Pressurizer empties	----
1,010.2	Safety injection actuation signal, psia	1,560
1,020.9	Low steam generator level signal, lbs (27.027 ft. above tubesheet)	134,540
1,073.8	Auxiliary feedwater flow initiated	----
1,305.6	No. 2 steam generator safety valves close, psia	926[(a)]
1,305.6	No. 1 steam generator safety valves close, psia	926[(a)]
1,800.0	Operator isolates damaged steam generator and opens atmospheric steam dump valve to the unaffected steam generator to begin plant cooldown to shutdown cooling	----
11,242.0	Shutdown cooling initiated, °F	350

(a) The original accident analysis assumed 4% main steam safety valve blowdown. Blowdown for current valve ring settings is bounded by 15% (see Section 15.0). The increased blowdown affects the MSSV steam releases during the first 30 minutes of the accident. The evaluation of radiological consequences was revised to include the effect of increased MSSV blowdown. The conclusions of the analysis remain valid. The doses are a small fraction of 10CFR100 exposure guidelines.

Table 15.6-6

ASSUMPTIONS FOR THE STEAM GENERATOR TUBE RUPTURE WITH CONCURRENT LOSS OF NONEMERGENCY AC POWER ANALYSIS

Parameter	Assumption
Initial core power level, MWt	3,478
Core inlet coolant temperature, °F	553
Core mass flowrate, 10^6 lbm/hr	154.3
Reactor coolant system pressure, psia	2,400
Steam generator pressure, psia	900
Moderator temperature coefficient, 10^{-4} $\Delta k/k/°F$	-3.3
Doppler coefficient multiplier	0.85
CEA worth for trip, $\%\Delta\rho$	-6.0
Steam bypass control system	Inoperative
Feedwater regulating system	Inoperative

5. Rapid drop in the volume control tank level

Based on this information, the operator can ramp down or manually trip the reactor if a trip has not occurred within the initial 30 minutes. In so doing, releases to the site boundary will be limited because of the relatively lower concentration in the steam generators.

The following assumptions and parameters are used to calculate the activity releases and offsite doses for a steam generator tube rupture (SGTR):

(a) The RCS equilibrium activity is based on long-term operation at 105% of the ultimate core power level of 3390 MWt (3390 MWt x 1.05 = 3560 MWt) with 1% failed fuel. These activities are given in table 11.1-2. See paragraph 15.6.3.2.5 items B and C for a discussion on iodine spikes.

(b) The steam generator equilibrium activity for both steam generators is assumed to be 0.1 μCi/g dose equivalent I-131 (Technical Specification limit) prior to the accident.

(c) Offsite power is lost; the main condenser is not available for steam relief via the turbine bypass system.

(d) Following the accident, no additional steam and radioactivity are released to the environment when the shutdown cooling system is placed in operation (3.12 hours).

(e) There is no main condenser evacuation system release and no steam generator blowdown during the accident.

(f) Only one steam generator is affected.

(g) The amount of noble gas activity released is equal to the amount present in the reactor coolant discharged into the secondary side following the tube rupture. The amount of noble gas activity contained in the secondary system is negligible in comparison.

(h) Iodine activity released is based on the equilibrium activity present in the steam generators (0.1 μCi/g dose equivalent I-131) and the amount of activity present in the reactor coolant discharged into the affected steam generator.

(i) Thirty minutes after the accident, the affected unit is isolated. No steam and fission product activities are released from the affected steam generator thereafter.

(j) The total amount of discharge of reactor coolant into the secondary system through the rupture is 75,672 pounds (in 30 minutes).

(k) The post-accident partition coefficient of 0.1 was used in the steam generator between the water and steam phases.

(1) The primary-to-secondary leakage of 8640 lbm/d (1.0 gal/min) is assumed to be applicable to the unaffected steam generator. The portion of the noble gas activity from the primary-to-secondary leakage attributed to the unaffected steam generator is assumed to be released during the course of the accident (3.12 hours).

(m) The amount of discharge of steam from the unaffected steam generators is calculated to be 6.33×10^5 pounds and, from the affected steam generator, 5.52×10^4 pounds.

(n) The activity released from the affected and unaffected steam generators is immediately vented to the atmosphere. The release point is assumed to be the safety valve nearest the control room air intake. No credit is taken for radioactive decay for isotopes in transit to dose points.

C. Results

The dynamic behavior of important NSSS parameters following a steam generator tube rupture with a concurrent loss of normal AC power are presented in figures 15.6-1 through 15.6-14.

The primary-to-secondary leakage due to a steam generator tube rupture results in a gradual decrease in the reactor coolant inventory and a gradual drop in the pressurizer level and pressure. As the level drops in the pressurizer, all charging pumps are brought on line while the letdown flow is reduced to a minimum. At approximately 15 minutes after initiation of the tube rupture, the reactor trips due to a low pressurizer pressure trip signal. The reactor trip results in the steam generator pressure rapidly increasing and opening the steam generator safety valves, since no credit is taken for the turbine bypass system. As a result of the low RCS pressure, a SIAS is also generated. As the safety injection system returns water to the RCS, the rate of RCS pressure drop decreases.

The primary system pressure and the pressurizer level in terms of water volume during the transient are shown as functions of time in figures 15.6-3 and 15.6-5, respectively. The steam generator pressure and steam generator levels in terms of liquid mass are shown as functions of time in figures 15.6-6 and 15.6-10, respectively.

On reactor trip, the loss of normal AC power is assumed to occur. As the reactor coolant pumps coast down, transfer of energy to the main steam system is reduced and the RCS pressure increases rapidly for approximately 60 psi before peaking out and commencing a more gradual drop. On reactor trip, the steam generator pressure rapidly increases, opening the steam generator safety valves. With the loss of feedwater, the steam generator pressure remains above the reseat pressure of the steam generator safety valves. The residual water inventory in the steam generators drops and initiates a low steam generator water level trip signal, which, in turn, automatically initiates an auxiliary feedwater flow signal. With the initiation of auxiliary feedwater flow, the steam generator pressure again drops gradually and the steam generator safety valve closes. Both the primary and secondary pressure continue to drop for the remainder of the transient.

After 30 minutes, the operator has identified the steam generator with the tube rupture based on information supplied relative to the steam generator water level and has closed the main steam isolation valve and stopped auxiliary feedwater flow to the damaged steam generator and terminated atmospheric releases from that steam generator. Plant cooldown is initiated by dumping steam from the intact steam generator. After the temperature of the reactor coolant is reduced to 350°F, the operator activates the shutdown cooling system and isolates both steam generators.

The maximum RCS and secondary pressures do not exceed 110% of design pressure following a steam generator tube rupture with concurrent loss of normal AC power, thus assuring that the integrity of the RCS and main steam system is not further degraded. The minimum DNBR of greater than 1.31 indicates no violation of the fuel thermal limits.

15.6.3.2.4 Barrier Performance

A. Mathematical Model

The mathematical model used for elevation of barrier performance is identical to that described in paragraph 15.6.3.2.3.

B. Input Parameters and Initial Conditions

The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.6.3.2.3.

C. Results

Figure 15.6-11 gives the steam generator safety valves flowrates versus time for the steam generator tube rupture with concurrent loss of normal AC power transient. At 30 minutes when the atmospheric steam dump valve is opened, the steam generator safety valves will have discharged no more than 101,830 pounds of steam; and during the first 30 minutes while the auxiliary feedwater pumps are operating, an additional 8510 pounds of steam will have been vented to the atmosphere via the steam-driven auxiliary feedwater pump. The operator then begins a 75°F/hr cooldown requiring a steam release rate of 61.3 lbm/s through the atmospheric dump valve and the auxiliary feedwater steam turbine. Approximately 578,210 pounds of steam would be discharged through the atmospheric steam dump valve and the auxiliary feedwater steam turbine during the 2.62-hour cooldown, giving a total steam release to the atmosphere of 688,550 pounds. For the first 2 hours, the combined steam release to the atmosphere is 440,120 pounds.

15.6.3.2.5 Radiological Consequences

A. Design Basis, Method of Analysis, No Iodine Spike

1. Physical Model

The evaluation of the radiological consequences of a postulated steam generator tube rupture assumes a complete severance of a single steam generator tube while the reactor is operating at full rated power and a coincident loss of offsite power at the time of reactor trip. Occurrence of the accident leads to an increase in contamination of the secondary system due to reactor coolant leakage through the tube break. A reactor trip occurs automatically as a result of low pressurizer pressure at approximately 985 seconds after the tube rupture occurs. The reactor trip automatically trips the turbine.

The resulting increase in radioactivity in the secondary system is detected by radiation monitors (refer to section 11.5). The coincident loss of offsite station power causes closure of the turbine bypass valves to protect the condenser. The steam generator pressure will increase rapidly, resulting in steam discharge as well as activity release through the main steam safety valves. Venting from the affected steam generator; i.e., the steam generator, which experiences tube rupture, continues until the secondary steam pressure is below the main steam safety valve setpoint. At this time, the affected steam generator is effectively isolated, and, thereafter, no steam or activity is assumed to be released from the affected steam generator. The remaining unaffected steam generator removes core decay heat by venting steam through the atmospheric dump valve and steam-driven auxiliary turbine until cooldown can be accomplished with the shutdown cooling system.

The analysis of the radiological consequences of a steam generator tube rupture considers the most severe release of secondary activity as well as reactor activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment is a function of the primary-to-secondary coolant leakage rate, the percentage of defective fuel in the core, and the mass of steam discharged to the environment. Conservative assumptions are made for all these parameters.

The sequence of events for this accident is presented in Table 15.6-5.

2. Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Table 15.6-6.

3. Mathematical Models Used in the Analysis

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

(a) The mathematical model used to analyze the activity released during the course of the accident is described in appendix 15B.

(b) The atmospheric dispersion factors used in the analysis, which are based on meteorological conditions assumed present during the course of the accident, are calculated according to the model described in subsection 2.3A. For the design basis analysis, the 5% level χ/Qs presented in table 15B-4 were used.

(c) The potential thyroid inhalation dose and beta-skin and total-body gamma immersion dose to an individual exposed at the exclusion area boundary or outer boundary of the low population zone (LPZ) are analyzed using the models described in appendix 15B.

(d) The buildup of activity in the control room and the potential integrated dose to control room personnel are analyzed based on models described in appendix 15B.

4. Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences of a postulated steam generator tube rupture, the activity released from the affected steam generator is assumed to be released directly to the environment by the safety valves until the steam generator is isolated by the operator 30 minutes after the initiation of the accident. The activity released from the unaffected steam generator is from the safety valves until the safety valves shut, and then from the auxiliary feedwater pump turbine and atmospheric dump valve during the cooldown phase until the shutdown cooling system is placed in operation. Since the activity is released directly to the environment with no credit for plateout, retention, or decay, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated steam generator tube rupture.

5. Identification of Uncertainties and Conservatisms in the Evaluation of the Results

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a steam generator tube rupture are as follows:

(a) Reactor coolant equilibrium activities are based on 1% failed fuel, which is greater by a factor of two to eight than that normally observed in past pressurized water reactor (PWR) operation.

(b) Steam generator equilibrium activity for both steam generators is assumed to be equal to the Technical Specification limit. The Technical Specification limits are conservatively derived based on accidents such as the SGTR.

(c) Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is one of minor leaks of undetermined origin. Activity in the secondary steam system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceed the limits established in the Technical Specifications would lead to reactor shutdown. Therefore, it is unlikely that the total amount of activity considered available for release in this analysis would ever be realized.

(d) The coincident loss of offsite power with the occurrence of the reactor trip following the steam generator tube rupture is a conservative assumption. In the event of availability of offsite power, the turbine bypass valves will open, relieving steam to the main condenser. This will reduce the amount of steam and entrained activity discharged directly to the environment from the unaffected steam generators.

(e) The meteorological conditions assumed to be present at the site during the course of the accident are based on χ/Q values which are expected to be worse 5% of the time. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

(f) A conservative steam generator partition coefficient (PC) of 0.1 is used in the cooldown phase (release to atmospheric dump valve).

B. Design Basis, Coincident (pre-existing) Iodine Spike and SGTR

In this evaluation, a case of coincident iodine spike which already exists due to a previous power transient was considered. The mathematical models, assumptions, and parameters are described in paragraph 15.6.3.2.5, item A with the following exception:

The RCS inventory was assumed to be 60 μCi/g dose equivalent Iodine 131 vice the reactor coolant inventory shown in table 11.1-2 which is based on 105% of design core power and 1% failed fuel. This 60 μCi/g is the Technical Specification limit for full power operation following an iodine spike for up to 48 hours.

C. Design Basis, Spike Caused by the SGTR

In this evaluation, a case with an iodine spike which was caused by the reactor trip following the SGTR was evaluated for radiological consequences. The mathematical models, assumptions, and parameters used are described in paragraph 15.6.3.2.5 with the following exception:

Prior to the SGTR, the RCS activity is based on 105% of design power and 1% failed fuel. This reactor coolant inventory is the same as that used in paragraph 15.6.3.2.5. However, at the initiation of the SGTR accident, the I-131 equivalent source term (released from fuel) is assumed to increase as discussed in paragraph 15.1.3.1B.5.3.

D. Realistic Analysis, Method of Analysis

A steam generator tube rupture (SGTR) is classified as a limiting fault. This accident is not expected to occur during the life of the plant but is postulated because the consequences of a SGTR include the potential for the release of significant amounts of radioactive materials. The term "realistic analysis" as used in this section does not imply that the accident is expected to occur during the life of the plant. The term "realistic analysis" signifies that more realistic assumptions and parameters have been used to evaluate the radiological consequences of a limiting fault as defined by Revision 2 of Regulatory Guide 1.70. Major assumptions and parameters used in the realistic analysis are presented in Table 15.6-8. The radiological consequences are presented in Table 15.6-7.

Table 15.6-7

RADIOLOGICAL CONSEQUENCES OF A POSTULATED STEAM GENERATOR TUBE RUPTURE W/LOAC (Sheet 1 of 2)

Results	Design Basis Assumptions	Realistic Assumptions
Exclusion Area Boundary Dose (0 to 2-hour), rem		
No iodine spike		
Thyroid	3.4	3.9×10^{-4}
Beta-skin	11.8×10^{-2}	8.2×10^{-5}
Total-body gamma	7.0×10^{-2}	4.5×10^{-5}
Coincident (pre-existing) iodine spike		
Thyroid	29.4	No spike
Beta-skin	12.8×10^{-2}	No spike
Total-body gamma	9.7×10^{-2}	No spike
Iodine spike caused by accident		
Thyroid	18.4	No spike
Beta-skin	12.4×10^{-2}	No spike
Total-body gamma	8.5×10^{-2}	No spike
LPZ Outer Boundary Dose (duration), rem		
No iodine spike		
Thyroid	10.2×10^{-2}	10.1×10^{-5}
Beta-skin	3.4×10^{-3}	2.1×10^{-5}
Total-body gamma	2.0×10^{-3}	11.7×10^{-6}

Table 15.6-7

RADIOLOGICAL CONSEQUENCES OF A POSTULATED STEAM GENERATOR TUBE RUPTURE W/LOAC (Sheet 2 of 2)

Results	Design Basis Assumptions	Realistic Assumptions
Control Room Dose (duration), rem		
No iodine spike		
Radiation External to the Control Room		
Total-body gamma	5.3×10^{-4}	6.0×10^{-6}
Radiation Internal to the Control Room		
Thyroid	1.2×10^{-2}	2.5×10^{-5}
Beta-skin	10.1×10^{-1}	1.4×10^{-2}
Total-body gamma	2.2×10^{-2}	2.9×10^{-4}

A realistic analysis of the radiological consequences of a postulated SGTR was performed. This analysis is identical with the evaluation presented in paragraph 15.6.3.2.5 with the following exceptions:

1. Reactor coolant system inventory is based on 0.12% failed fuel vice 1% failed fuel and 100% (3390 MWt) via 105% (3560 MWt) of the ultimate core power level respectively. Isotopic inventory is presented in table 11.1-3.

2. An iodine spike, pre-existing or caused by the accident, does not occur.

3. Steam generator equilibrium activity prior to the accident is based on 100 lbm/d and 0.12% failed fuel versus the Technical Specification limit for steam generator activity. Steam generator activity is presented in table 11.1-21 (normal case).

4. 50% level χ/Qs are used instead of 5% level χ/Qs.

5. A post-accident partition coefficient of 0.01 was used between the water and steam phases versus 0.1 for the design basis case.

Table 15.6-8

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE, W/LOAC (Sheet 1 of 4)

Parameter	Design Basis Assumption	Realistic Assumption
Source Data		
Power level, MWt	3,560	3,390
Fraction failed fuel, %	1	0.12
Steam generator tube leakage, lbm/d	8,640 (1 gal/min)	100
Table of equilibrium reactor coolant activity		
1. No iodine spike	Table 11.1-2	Table 11.1-3
2. Coincident (pre-existing) iodine spike μCi/g dose equivalent I-131	60	No spike
3. Iodine spike caused by accident	Section 15.1.3.1B.5.3[(a)]	No spike
Table of equilibrium secondary system activity	0.1 μCi/g dose equivalent I-131 (technical specification limit)	Table 11.1-2 (average case)
Activity Release Data		
Steam discharge, lb		
Affected steam generator		
Reactor coolant leakage to steam generator (0 to 30 min)	75,672	75,672
Mass of steam released	5.52×10^4	5.52×10^4

(a) Prior to accident, reactor coolant activity assumed to be based on 1.0% failed fuel (table 11.1-2). Following SGTR, activity assumed to increase as discussed in section 15.1.3.1B.5.3. Iodine release terms increase by a factor of 500.

Table 15.6-8

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE, W/LOAC (Sheet 2 of 4)

Parameter	Design Basis Assumption		Realistic Assumption	
Unaffected steam generator Duration of accident (3.12 hours)	6.33×10^5		6.33×10^5	
Iodine partition coefficients for steam generators (between water and steam phase)	0.1		0.01	
Activity released from steam generators, Ci				
No iodine spike				
Isotope	0-2 hour	Duration	0.2 hour	Duration
I-131	1.8(+1)	1.9(+1)	1.6(-1)	1.6(-1)
I-132	4.8(0)	5.0(0)	3.8(-2)	3.8(-2)
I-133	2.2(+1)	2.3(+1)	1.8(-1)	1.8(-1)
I-134	2.0(0)	2.0(0)	1.7(-2)	1.7(-2)
I-135	9.3(0)	9.6(0)	7.8(-2)	7.9(-2)
Xe-131m	8.3(+1)	8.4(+1)	3.8(0)	3.8(0)
Xe-133m	0	0	0	0
Xe-133	11.6(+3)	11.7(3)	6.2(+2)	6.3(+2)
Xe-135m	3.8(+1)	3.8(+1)	4.5(-1)	4.5(-1)
Xe-135	3.2(+2)	3.2(+2)	12.2(0)	12.2(0)
Xe-137	0	0	0	0
Xe-138	1.9(+1)	2.0(+1)	1.5(0)	1.5(0)
Kr-83m	0	0	0	0
Kr-85m	8.2(+1)	8.2(+1)	7.6(0)	7.7(0)
Kr-85	1.8(+2)	1.8(+2)	5.2(0)	5.2(0)
Kr-87	4.4(+1)	4.4(+1)	2.1(0)	2.1(0)
Kr-88	1.4(+2)	1.4(+2)	6.9(0)	7.0(0)

Table 15.6-8

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE, W/LOAC (Sheet 3 of 4)

Parameter	Design Basis Assumption	Realistic Assumption
Coincident (Pre-existing) iodine spike		
Isotope	0 to 2 hours	No iodine spike
I-131	1.5(+2)	No iodine spike
I-132	4.2(+1)	No iodine spike
I-133	1.9(+2)	No iodine spike
I-134	1.8(+1)	No iodine spike
I-135	8.3(+1)	No iodine spike
Xe-131m	8.3(+1)	No iodine spike
Xe-133m	0	No iodine spike
Xe-133	11.6(+3)	No iodine spike
Xe-135m	3.8(+1)	No iodine spike
Xe-135	3.2(+2)	No iodine spike
Xe-137	0	No iodine spike
Xe-138	1.9(+1)	No iodine spike
Kr-83m	0	No iodine spike
Kr-85m	8.2(+1)	No iodine spike
Kr-85	1.8(+2)	No iodine spike
Kr-87	4.4(+1)	No iodine spike
Kr-88	1.4(+2)	No iodine spike
Iodine spike caused by accident		
Isotope	0 to 2 hours	
I-131	9.4(+1)	No iodine spike
I-132	2.6(+1)	No iodine spike
I-133	11.7(+1)	No iodine spike
I-134	10.6(0)	No iodine spike
I-135	5.1(+1)	No iodine spike
Xe-131m	8.3(+1)	No iodine spike
Xe-133m	0	No iodine spike
Xe-133	11.6(+3)	No iodine spike
Xe-135m	3.8(+1)	No iodine spike
Xe-135	3.2(+2)	No iodine spike
Xe-137	0	No iodine spike
Xe-138	1.9(+1)	No iodine spike

Table 15.6-8

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE, W/LOAC (Sheet 4 of 4)

Parameter	Design Basis Assumption	Realistic Assumption
Isotope (continued)	0 to 2 hours	
Kr-83m	0	No iodine spike
Kr-85m	8.2(+1)	No iodine spike
Kr-85	1.8(+2)	No iodine spike
Kr-87	4.4(+1)	No iodine spike
Kr-88	1.4(+2)	No iodine spike
Dispersion data		
Distance to exclusion area boundary, meters	576	576
Distance to LPZ outer boundary, meters	3,140	3,140
Atmospheric dispersion factors, s/m^3	5% level χ/Q (table 15B-4)	50% level χ/Q (table 15B-4)
Control room design parameters	Refer to table 15B-5	Refer to table 15B-5

Table 15.6-9
(Deleted)

Table 15.6-10
(Deleted)

15.6.3.2.6 Conclusions

A. Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the SGTR is the control room filtration system. Activity loadings on the control room carbon adsorber are based on flowrate through the filter, concentration of activity at the filter inlet, and filter efficiency.

Activity loading on the control room carbon adsorber has been designed for the LOCA, paragraph 15.6.3.3.5.1. Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated steam generator tube rupture accident releases.

B. Dose to an Individual Exposed at the Exclusion Area Boundary and the Outer Boundary of the Low Population Zone.

The potential radiological consequences resulting from the occurrence of a postulated steam generator tube rupture were analyzed, using assumptions and models described in preceding subsections for both the design basis and realistic analyses.

The direct beta-skin and total-body gamma dose due to immersion and the thyroid dose due to inhalation were analyzed for the 0 to 2-hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results are listed in Table 15.6-8. The resultant doses are a small fraction of 10CFR100 for the design basis evaluation without iodine spike and are within 10CFR100 limits for cases considering an iodine spike.

C. Dose to Control Room Personnel

Radiation doses to control room personnel following a postulated steam generator tube rupture are based on the same shielding, ventilation, cavity dilution, and dose model assumptions as those given in appendix 15B. Buildup of activity in the control room is based on the 3.12-hour release of activity following the accident. Control room personnel are subject to total-body gamma dose due to immersion in the cloud concentrations internal to the control room. The thyroid inhalation dose is based on immersion in cloud concentrations internal to the control room.

The thyroid, beta-skin, and total-body gamma doses to control room personnel are listed in Table 15.6-8. The resultant doses are within the limits of 10CFR50, Appendix A, General Design Criterion 19.

15.6.3.2.7 RCS Voiding

The consequences due to potential void formation in the RCS during design basis transients are discussed in reference 2. The conclusions are that the void formation, if any, in the RCS is not great enough to impair reactor coolant circulation or core coolability, and that the impact of RCS voiding will not result in violation of NRC Standard Review Plan acceptance criteria. The conclusions reached in reference 2 are valid for the non-LOCA events, namely the steam generator tube rupture and the letdown line break presented in this section.

15.6.3.3 <u>Loss-of-Coolant Accident (LOCA)</u>

San Onofre Nuclear Generating Station, Unit 2 & 3

Updated Final Safety Analysis Report

Revised April 2011

Chapter 15.0 – Accident Analyses

Section 15.10.0 – Transient Analyses

15 ACCIDENT ANALYSIS

15.10 UPDATED ACCIDENT ANALYSES INCORPORATING 10 CFR 50.59 CHANGES

15.10.0 TRANSIENT ANALYSES

This chapter presents analytical evaluation of the response of the plant to postulated disturbances in process variables and to postulated malfunctions of 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 in accordance with references 34, 35, and 36 to determine the 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 subsection 15.0.1. The frequency of each incident[a] was estimated, and each incident was placed in one of three frequency categories specified in reference 1 and discussed in subsection 15.0.1.

In addition, a miscellaneous events category is established in section 15.9. This category was established specifically to include the Asymmetric Steam Generator Transient (ASGT). The ASGT does not conveniently fit into any of the other categories and was not specified by reference 1.

Section 15.10 was added in Revision 13 to present updated fuel cycle and unit specific data and consequences for the events presented in Sections 15.1 through 15.9. The contents of 15.10.1 through 15.10.9 are outlined in 15.10.0. When comparing 15.10 to 15.1 through 15.9 the following should be noted:

(a) Sections 15.1 through 15.9 are consistent with the latest information that has been reviewed and approved by the NRC. These sections are intended to be updated when information has been submitted to and approved by the NRC. The information in these sections is based upon a rated core thermal power of 3390 MWt, unless otherwise noted.

(b) Section 15.10 presents the current plant configuration. This section includes data that has been added through 10 CFR 50.59's since the last approval of the event information by the NRC. This section is intended to be updated under the requirements of 10 CFR 50.59. The information in this section is based upon a rated core thermal power of 3438 MWt.

San Onofre Nuclear Generating Station, Unit 2 & 3

Updated Final Safety Analysis Report

Revised April 2011

Chapter 15.0 – Accident Analyses

Section 15.10.6.3.2 – Steam Generator Tube Rupture With Concurrent Loss of AC Power

The offsite radiological doses for the Primary Sample or Instrument Line Break with an accident-induced iodine spike are "a small fraction" (i.e., do not exceed 10%) of the 10 CFR 100 exposure guidelines, and the Control Room radiological doses are within the 10 CFR 50 Appendix A General Design Criterion 19 exposure guidelines.

15.10.6.3.2 Steam Generator Tube Rupture with Concurrent Loss of AC Power

Introduction

A Steam Generator Tube Rupture (SGTR) event is a penetration of the barrier between the Reactor Coolant System (RCS) and the main steam system via the double-ended break of a U-tube. This causes highly radioactive RCS fluid to contaminate the secondary side. The radioactivity is released via the condenser air ejectors, the Main Steam Safety Valves (MSSVs), and the Atmospheric Dump Valves (ADVs).

This event is analyzed with a concurrent loss of AC power, which increases the radiological release to the environment (see section 15.6.3.2.5). It is this analysis which is presented below.

If the primary to secondary leak is beyond the capacity of the charging pumps, the reactor will eventually trip on a low pressure trip signal. As a result of the loss of AC, the electrical power would be unavailable for the station auxiliaries such as the Reactor Coolant Pumps (RCPs) and the Main Feed Water (MFW) pumps. Under such circumstances, the plant would experience a simultaneous loss of load, normal feed water flow, forced reactor coolant flow and steam generator blowdown capability.

When the reactor is off line, stored energy and fission product decay energy must be dissipated by the reactor coolant and main steam systems. In the absence of forced reactor coolant flow, convective heat transfer is supported by natural circulation reactor coolant flow. Initially, the liquid inventory in the steam generators is used and the resultant steam is released to the atmosphere via the MSSVs. With the availability of stand-by power provided by the automatic start-up of the diesel generators, Auxiliary (emergency) Feed Water (AFW) flow is initiated on a low steam generator level signal.

When the reactor plant has been stabilized in Mode 3, the operator achieves plant cool down using remotely operated ADVs. The plant is cooled to 350°F at a nominal rate of 75°F/hr. At this time, Shut Down Cooling (SDC) is initiated.

The analysis of record conservatively assumes the operator action to isolate the affected steam generator is delayed until 30 minutes after initiation of the event. The operator's diagnosis of the SGTR event is facilitated by the radiation monitors which initiate alarms and signal the existence of abnormal radioactivity levels.

Radiation monitors are found in the blowdown sample lines from each steam generator, in the blowdown processing system neutralization sump discharge sea line which processes blowdown from both steam generators, and in the condenser air ejector discharge line. Additional diagnostic information is provided by RCS pressure and pressurizer level response indicating a loss of primary coolant. Level in the affected steam generator increases as the primary fluid enters the steam generator driven by the substantially higher primary pressure.

The offsite and control room dose consequences of the postulated steam generator tube rupture are analyzed for the assumed conditions of no iodine spike, a pre-accident iodine spike, and an accident initiated iodine spike in the reactor coolant.

Summary of Methods

The CESEC-III code is used to simulate the transient for the first 1800 seconds (i.e., 30 minutes). The output of the code provides the amount of primary to secondary leak, the amount of steam transported from the steam generators through the MSSVs and the overall Nuclear Steam Supply System (NSSS) response to the event. This information is then used to derive the radiological releases and accompanying doses.

This analysis is primarily performed to establish the parameters, such as the primary to secondary mass transferred during the event, by which the radiological releases are calculated. There is no specific acceptance criteria for the mass releases.

One computer case was run for this analysis. This was an 1800 seconds CESEC-III simulation of a double-ended SGTR in the right hand (arbitrary designation) steam generator. This case utilizes a 15% MSSV blowdown model to determine the impact on steam released to atmosphere.

The primary transient analysis inputs and assumptions for the analysis are presented below in Table 15.10.6.3.2-1. The sequence of events is provided in Table 15.10.6.3.2-2.

The dose methodology for this event is described in Appendices 15B and 15.10.B. Using this methodology, design basis 0-2 hour Exclusion Area Boundary, 0-30 day Low Population Zone, and 0-30 day Control Room doses were calculated with and without consideration of pre-existing and accident induced iodine spikes.

The following release mechanisms that can disperse radioactive material into the atmosphere have been evaluated:

1. Reactor coolant releases via the ruptured tube into the affected steam generator, and eventually to the outside environment.

2. Normal primary to secondary leakage releases into the affected and intact steam generators, and eventually to the outside environment.

3. Main steam safety valve releases from the affected and intact steam generators to the outside environment.

4. Turbine-driven auxiliary feed water pump venting of secondary steam from the affected and intact steam generators to the outside environment.

5. Atmospheric dump valve releases of secondary steam from the intact steam generator to the outside environment.

6. Leakage past one or more of the affected steam generator MSSVs and/or its ADV (subsequent to operator action to isolate the affected steam generator). Conservatively, the total leakage is modeled as being equivalent to the flow capacity of a MSSV.

The principal assumptions and inputs for the dose analysis are presented below in Table 15.10.6.3.2-3.

Table 15.10.6.3.2-1
Principal Assumptions and Inputs for SGTR

Parameter	Unit 2	Unit 3
Core Power	3478 MWth	3478 MWth
Inlet Temperature	560°F	560°F
RCS Pressure	2300 psia	2300 psia
SG Pressure	900 psia	900 psia
Core Flow, Total	376,200 gpm	376,200 gpm
BOC Doppler Uncertainty Multiplier	0.86	0.86
Moderator Temperature Coefficient	$-3.7 \times 10^{-4}\ \Delta\rho/°F$	$-3.7 \times 10^{-4}\ \Delta\rho/°F$
SCRAM Worth	-6.0 % $\Delta\rho$	-6.0 % $\Delta\rho$
CPC (range - low pressure) Trip Set Point	1785 psia	1785 psia
Loss of AC Power	Coincident with Reactor Trip	Coincident with Reactor Trip
Steam Generator (S/G) U-Tube Break Size	45% Double Ended Guillotine	45% Double Ended Guillotine
Safety Injection Actuation System - Set Point	1785 psia	1785 psia
High Pressure Safety Injection - Response Time	15.0 seconds	15.0 seconds
Main Feed Water (MFW) - Flow Rate	102% of Design	102% of Design
Main Feed Water (MFW) – Enthalpy	425 Btu/lbm (pre-trip)	425 Btu/lbm (pre-trip)
Auxiliary Feed Water (AFW) - Response Time	57.7 secs. electric and steam driven	57.7 secs. electric and steam driven
Auxiliary Feed Water (AFW) - Flow Rate	601 gpm at 1000 psia	601 gpm at 1000 psia
Auxiliary Feed Water (AFW) - Enthalpy	68 Btu/lbm	68 Btu/lbm
Charging Flow Rate	135 gpm	135 gpm
Letdown Flow Rate	0.0 gpm	0.0 gpm

Table 15.10.6.3.2-1 (continued)

Principle Assumptions and Inputs for SGTR

Parameter	Unit 2	Unit 3
Main Steam Safety Valves (MSSV) - Opening Set Points (9 valves at 7 psi increments. Includes set point tolerance)	1067 to 1120.4 psia -3% Tolerance	1067 to 1120.4 psia -3% Tolerance
MSSV Accumulation Set Point	0%	0%
MSSV Blow Down	15% (to fully close)	15% (to fully close)
Atmospheric Dump Valves (ADVs)	Inoperative	Inoperative
Feed Water Control System (FWCS)	Not Required to Mitigate Event	Not Required to Mitigate Event
Pressurizer Pressure Control System (PPCS)	Not Required to Mitigate Event	Not Required to Mitigate Event
Steam Bypass Control System (SBCS)	Inoperative	Inoperative

Table 15.10.6.3.2-2a
Sequence of Events for SGTR, Unit 2

Time (Seconds)	Chronological Event	Set Point or Value
0.0	S/G Tube rupture occurs	----
1329.4	CPC Reactor Trip (low pressurizer pressure) setpoint reached, SIAS initiated	1785 psia
1330.3	Reactor Trip breakers open Turbine Stop Valves close Loss of Normal AC	----
1333.1	MSSVs begin to open on both S/Gs	1067 psia
1337.9	Maximum S/G pressure on both generators	1112 psia
1356.1	Low S/G level signal generated, AFW initiated	115,610 lbm
1795.8	MSSVs close on both S/Gs	907 psia
1800.0	Damaged S/G isolated, ADV on unaffected S/G opened to begin system cool down to Shut Down Cooling (SDC)	----
11880.0	Shut Down Cooling (SDC) initiated Temperature Total steam release	 350°F 739,034 lbm

Table 15.10.6.3.2-2b

Sequence of Events for SGTR, Unit 3

Time (Seconds)	Chronological Event	Set Point or Value
0.0	S/G Tube rupture occurs	----
1329.4	CPC Reactor Trip (low pressurizer pressure) setpoint reached, SIAS initiated	1785 psia
1330.3	Reactor Trip breakers open Turbine Stop Valves close Loss of Normal AC	----
1333.1	MSSVs begin to open on both S/Gs	1067 psia
1337.9	Maximum S/G pressure on both generators	1112 psia
1356.1	Low S/G level signal generated, AFW initiated	115,610 lbm
1795.8	MSSVs close on both S/Gs	907 psia
1800.0	Damaged S/G isolated, ADV on unaffected S/G opened to begin system cool down to Shut Down Cooling (SDC)	----
11880.0	Shut Down Cooling (SDC) initiated Temperature Total steam release	 350°F 739,034 lbm

Table 15.10.6.3.2-3

Principal Assumptions and Inputs for Steam Generator Tube Rupture Dose Analysis

Parameter	Unit 2	Unit 3
AC Availability	Loss of AC Power	Loss of AC Power
RCS Iodine Activity (Dose Equivalent I-131), μCi/gm	1.0	1.0
Increase in Iodine Release Rate from Fuel for Accident Induced Iodine Spike	500	500
RCS Pre-Existing Iodine Spike Iodine Activity (Dose Equivalent I-131), μCi/gm	60	60
RCS Non-Iodine Activity, μCi/gm	100/□	100/□
Secondary Liquid Iodine Activity (Dose Equivalent I-131), μCi/gm	0.1	0.1
Steam Generator Iodine Partition Coefficient	0.01	0.01
Primary to Secondary Leak Rate into each SG, gpm	0.5	0.5
Integrated primary to secondary rupture flow, lbm (1,800 seconds)	70,563	70,563
Additional primary to secondary rupture and normal leakage flow available for release from 30 minutes to shut down cooling due to MSSV/ADV valve seat leakage, lbm	150,000	150,000
Integrated MSSV flow, lbm (1,800 seconds) LH – Unaffected RH – Affected Total MSSV Flow	 57,560 57,664 115,224	 57,560 57,664 115,224
AFW Flow (steam driven pump), lbm (1,800 seconds)	4,922	4,922
Steam Release (30 – 120 minutes), lbm	331,547	331,547
Total steam release (0 - 120 minutes), lbm	451,693	451,693
Total steam release to Shut Down Cooling, lbm	739,034	739,034
Additional affected steam generator steam release from 30 minutes to shut down cooling due to MSSV/ADV valve seat leakage, lbm	2,400,000	2,400,000
Control Room Isolation Signal	High Radiation	High Radiation
Control Room Isolation Time, min	3	3
Offsite Dose Evaluation Model	Appendix 15B and Appendix 15.10B	Appendix 15B and Appendix 15.10B
Control Room Dose Evaluation Model	Appendix 15B and Appendix 15.10B	Appendix 15B and Appendix 15.10B

Results

The primary to secondary mass transfer and steam release data required to perform radiological calculations for the steam generator tube rupture event are presented in Table 15.10.6.3.2-3.

The RCS and secondary system pressures remain below the 110% of the design pressure limits, thus, assuring the integrity of these systems.

The results of the most recent analysis of the potential off site and control room personnel doses from a steam generator tube rupture with concurrent loss of normal AC power are presented in Table 15.10.6.3.2-4. These results are compared against the NRC approved acceptance criteria in section 15.6.3.2.

Table 15.10.6.3.2-4

Results for Steam Generator Tube Rupture

Parameter	Acceptance Criteria	Analysis Results	
		Unit 2	Unit 3
Design Basis Case with No Iodine Spike			
0-2 hr EAB Doses, Rem			
Thyroid	30	0.8	0.8
Beta Skin	N/A	0.1	0.1
Whole Body	2.5	0.2	0.2
0-30 day LPZ Doses, Rem			
Thyroid	30	<0.1	<0.1
Beta Skin	N/A	<0.1	<0.1
Whole Body	2.5	<0.1	<0.1
0-30 day Control Room Doses, Rem			
Thyroid	30	1.8	1.8
Beta Skin	30	1.6	1.6
Whole Body	5	<0.1	<0.1
Design Basis Case with Pre-Existing Iodine Spike			
0-2 hr EAB Doses, Rem			
Thyroid	300	8.2	8.2
Beta Skin	N/A	0.1	0.1
Whole Body	25	0.2	0.2
0-30 day LPZ Doses, Rem			
Thyroid	300	0.2	0.2
Beta Skin	N/A	<0.1	<0.1
Whole Body	25	<0.1	<0.1
0-30 day Control Room Doses, Rem			
Thyroid	30	2.0	2.0
Beta Skin	30	1.6	1.6
Whole Body	5	<0.1	<0.1
Design Basis Case with Accident Induced Iodine Spike			
0-2 hr EAB Doses, Rem			
Thyroid	30	4.1	4.1
Beta Skin	N/A	0.1	0.1
Whole Body	2.5	0.2	0.2
0-30 day LPZ Doses, Rem			
Thyroid	30	0.1	0.1
Beta Skin	N/A	<0.1	<0.1
Whole Body	2.5	<0.1	<0.1
0-30 day Control Room Doses, Rem			
Thyroid	30	1.9	1.9
Beta Skin	30	1.6	1.6
Whole Body	5	<0.1	<0.1