15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident is described in WCAP-15806-P- A~~7588, Revision 1A~~ (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

15.4.8.1.2 Limiting Criteria

This event is a Condition IV incident (ANSI N18.2). See subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

~~Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project (Reference 5). Extensive tests of uranium dioxide (UO2) zirconium-clad fuel rods representative of those in pressurized water reactor cores such as AP1000 have demonstrated failure thresholds in the range of 240 to 257 cal/g. Other rods of a slightly different design have exhibited failure as low as 225 cal/g. These results differ significantly from the TREAT (Reference 6) results, which indicated a failure threshold of 280 cal/g. Limited results indicate that this threshold decreases by about 10 percent with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods.~~

~~Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/g for unirradiated rods and 200 cal/g for irradiated rods. Catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/g.~~

~~Regulatory Guide 1.77~~ NUREG-0800 Standard Review Plan (SRP) 4.2, Revision 3 (Reference 24), interim criteria applicable to new plant design certification are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure ~~B~~ B-1 of SRP 4.2, Revision 3, Appendix B.
- The high cladding temperature failure criteria for ~~zero~~ zero-power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.
- For intermediate (greater than ~~5~~ 5-percent rated thermal power) and ~~full~~ full-power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g., DNBR).
- For core coolability, it is conservatively assumed that the average~~Average~~ fuel pellet enthalpy at the hot spot remains~~is~~ below 200 cal/g (360 ~~b~~Btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.

- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten ~~fuel~~ fuel-~~to~~-to-coolant interaction and (2) fuel rod burst that must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.
- Peak reactor coolant system pressure is less than that which could cause stresses to exceed the "Service Limit C" as defined in the ASME code.
- ~~Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of the first criterion~~.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core ~~channel~~ calculation and then, a hot ~~regi~~o~~n~~d calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy, ~~and~~fuel temperature, and DNB transients ~~at the hot spot~~ are then determined by ~~multiplying the average core energy generation by the hot channel factor and~~ performing a conservative fuel rod transient heat transfer calculation. ~~The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.~~

A discussion of the method of analysis appears in WCAP-~~7588, Revision 1A~~15806-P-A (Reference 4).

Average Core Analysis

The ~~spatial kinetics computer~~three-dimensional nodal code ~~TWINKLE~~ANC (References 14, 15, 16, 17, 21, 22, and 27) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in ~~1, 2, or 3~~three spatial dimensions (rectangular coordinates) for ~~6~~six delayed neutron groups. The core moderator and fuel temperature feedbacks are based on the NRC approved Westinghouse version of the VIPRE-01 code and methods (References 18 and 19). ~~and up to 2000 spatial points. The computer code includes a multiregion, transient fuel-clad-coolant heat transfer model for the calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code because it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. Because the radial dimension is missing, it is necessary to use conservative methods (described as follows) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in subsection 15.0.11.~~

Hot ~~Spot~~Rod Analysis

The hot fuel rod models are based on the Westinghouse VIPRE models described in WCAP-15806-P-A (Reference 4). The hot rod model represents the hottest fuel rod from any channel in the core. VIPRE performs the hot rod transients for fuel enthalpy, temperature, and DNBR using as input the time dependent nuclear core power and power distribution from the core average analysis. A description of the VIPRE code is provided in Reference 18. ~~In the hot spot analysis, the initial heat flux is equal to the nominal value multiplied by the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. The assumption is made that the hot spots before and after ejection are coincident. This is conservative because the peak after ejection occurs in or adjacent to the assembly with the ejected rod, and before ejection, the power in this region is depressed.~~

~~The hot spot analysis is performed using the fuel and cladding transient heat transfer computer code FACTRAN (Reference 2). This computer code calculates the transient temperature distribution in a cross section of a metal-clad UO2 fuel rod and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium water reaction is explicitly represented, and material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.~~

~~FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB and the Bishop-Sandburg-Tong correlation (Reference 8) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used, assuming zero-bulk fluid quality. The DNBR is not calculated. Instead, the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient is calculated by the code. It is adjusted to force the full power, steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in subsection 15.0.11.~~

System Overpressure Analysis

~~There is little likelihood of fuel dispersal into the coolant. The pressure surge may be calculated on the basis of conventional heat transfer from the fuel and prompt heat absorption by the coolant~~If the fuel coolability limits are not exceeded, the fuel dispersal into the coolant or a sudden pressure increase from thermal to kinetic energy conversion is not needed to be considered in the overpressure analysis. Therefore, the overpressure condition may be calculated on the basis of conventional fuel rod to coolant heat transfer and the prompt heat generation in the coolant. The system overpressure analysis is conducted by first performing the core power response analysis to obtain the nuclear power transient (versus time) data. The nuclear power data is then used as input to a plant transient computer code to calculate the peak reactor coolant system pressure.

~~The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a (Section 4.4) calculation is performed to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code.~~ This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected~~on the basis of values calculated for this type of core. Table 15.4-3 presents the important parameters used in this analysis~~as described in Reference 4.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using ~~either~~ three-dimensional static methods ~~or by a synthesis method using one-dimensional and two dimensional calculations~~. Standard nuclear design codes are used in the analysis. ~~No credit is taken for the flux flattening effects of reactivity feedback.~~ The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate safety analysis ~~margins~~allowances are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification as discussed in Reference 4.

~~Power distributions before and after ejection for a worst case can be found in WCAP-7588, Revision 1A (Reference 4). During plant startup physics testing, rod worths and power distributions have been measured in the zero-power configuration and compared to values used in the analysis. The ejected rod worth and power peaking factors are consistently overpredicted in the analysis.~~

15.4.8.2.1.2 ~~Reactivity Feedback Weighting Factors~~Not Used

~~The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis.~~

~~Physics calculations are carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes are compared, and effective reactivity feedback weighting factors are shown to be conservative. These weighting factors take the form of multipliers that, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape.~~

~~In this analysis, because a one-dimensional (axial) spatial kinetics method is used, axial reactivity weighting is not necessary if the initial condition matches the ejected rod configuration. In addition, no reactivity weighting is applied to the moderator feedback.~~

~~A conservative radial reactivity weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time, accounting for the missing spatial dimension. These reactivity weighting factors are shown to be conservative compared to three dimensional analysis (Reference 5).~~

15.4.8.2.1.3 Moderator and Doppler Coefficients

The critical boron concentration~~s at the beginning of cycle and end of cycle are~~ is adjusted in the nuclear code to obtain a moderator ~~density~~temperature coefficient ~~curves~~ that ~~are~~is conservative compared to

actual design conditions for the plant consistent with Reference 4. The fuel temperature feedback in the neutronics code is reduced consistent with Reference 4 requirements. ~~No weighting factor is applied to these results.~~

~~The Doppler reactivity defect is determined as a function of power level using a one-dimensional, steady-state computer code with a Doppler weighting factor of one. The Doppler defect used is given in subsection 15.0.4. The Doppler weighting factor increases under accident conditions.~~

15.4.8.2.1.4 Delayed Neutron Fraction, βeff

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than ~~0.70 percent at beginning of cycle and~~ 0.50 percent at end of cycle ~~for the first cycle~~. The accident is sensitive to $\boldsymbol{\beta}_{eff}$ if the ejected rod worth is equal to or greater than $\boldsymbol{\beta}_{eff}$ ~~as in zero-power transients~~. To allow for future cycles, a pessimistic estimate~~s~~ of $\boldsymbol{\beta}_{eff}$ of ~~0.49 percent at beginning of cycle and~~ 0.44 percent ~~at end of cycle are~~is used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

The trip reactivity insertion ~~assumed is given in Table 15.4-3 and includes~~accounts for the effect of the ejected rod and one adjacent stuck ~~RCCA~~rod. ~~These values are reduced by the ejected rod reactivity.~~ The ~~shutdown~~trip reactivity is simulated by dropping a limited set of rods of the required worth into the core. The start of rod motion occurs 0.9 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.583 second for the instrument channel to produce a signal, 0.167 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.47 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

15.4.8.2.1.7 Results

For all cases, the core is preconditioned by assuming a fuel cycle depletion with control rod insertion that is conservative relative to expected baseload operation. All~~Because the control rod insertion limits for the AP1000 are multidimensional, a significant number of rodded configurations are evaluated to determine the most limiting cases, (that is, those cases that produced the least amount of margin to the Standard Review Plan Section 15.4.8 evaluation acceptance criteria). The hot zero power~~ cases ~~and hot full power cases~~ assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event and xenon is skewed to yield a conservative initial axial power shape. The limiting RCCA ejection cases~~,~~ for ~~both the beginning and end of~~a typical cycle ~~at zero and full power,~~ are ~~presented next~~summarized following the criteria outlined in subsection 15.4.8.1.2.

- PCMI and high cladding temperature (hot zero power)

  The resulting maximum fuel average enthalpy rise and maximum fuel average enthalpy are less than the criteria given in subsection 15.4.8.1.2.

- High cladding temperature ($\geq$ 5% rated thermal power)

The fraction of the core calculated to have a DNBR less than the safety analysis limit is less than the amount of failed fuel assumed in the dose analysis described in subsection 15.4.8.3.

- Core coolability

The resulting maximum fuel average enthalpy is less than the criterion given in subsection 15.4.8.1.2. Fuel melting is not predicted to occur at the hot spot.

There are no fuel failures due to the fuel enthalpy deposition, i.e., both fuel and cladding enthalpy limits were met. Additionally, the coolability criteria for peak fuel enthalpy and the fuel melting criteria were met. Therefore, the fuel dispersal into the coolant, a sudden pressure increase from thermal to kinetic energy conversion, gross lattice distortion, or severe shock waves are precluded.

- ~~Beginning of cycle, full power~~

~~The limiting ejected rod worth and hot channel factor are conservatively assumed to be 0.37 percent Δk and 4.9, respectively. The peak hot spot cladding average temperature is 2265°F. The peak hot spot fuel center temperature reaches melting at 4900°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.~~

- ~~Beginning of cycle, zero power~~

~~For this condition, the limiting ejected rod worth and hot channel factor are conservatively assumed to be 0.65 percent Δk and 12.0, respectively. The peak hot spot cladding average temperature is 1907°F, and the peak hot spot fuel center temperature is 3018°F.~~

- ~~End of cycle, full power~~

~~The ejected rod worth and hot channel factor are conservatively assumed to be 0.30 percent Δk and 6.0, respectively. The peak hot spot cladding average temperature is 2151°F. The peak hot spot fuel temperature reaches melting at 4800°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.~~

- ~~End of cycle, zero power~~

~~The ejected rod worth and hot channel factor for this case are conservatively assumed to be 0.75 percent Δk and 19.6, respectively. The peak hot spot cladding average temperature is 2122°F, and the peak hot spot fuel center temperature is 3263°F.~~

~~A summary of the preceding cases is given in Table 15.4-3.~~ The nuclear power and fuel ~~and cladding temperature~~ transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-34.

The calculated sequence of events for the limiting cases ~~rod ejection accidents, as shown in Figures 15.4.8 1 through 15.4.8-4,~~ is presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 10 percent of the rods are assumed to enter DNB based on a detailed three-dimensional kinetics and hot rod analysis. The maximum fuel average enthalpy rise of rods predicted to enter DNB will be less than 60 cal/g. Fuel melting does not occur at the hot spot.~~THINC analysis (Reference 4). Although limited (less than 10 percent) fuel melting at the hot spot is allowed for the full-power cases, in practice, melting is not expected because the analysis conservatively assumes that the hot spots before and after ejection are coincident.~~

15.4.8.2.1.9 Peak Reactor Coolant System Pressure ~~Surge~~

~~A c~~Calculations of the ~~pressure surge~~peak reactor coolant system pressure~~for an ejection worth of about one dollar at beginning of cycle, hot full power,~~ demonstrate~~s~~ that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. ~~Because the severity of the analysis does not exceed the worst-case analysis~~Therefore, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

15.4.8.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with a limited number of fuel rods containing cladding defects~~the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects)~~ and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. See subsection 15.4.8.3.1 and Table 15.4-4.

As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see subsection 15.4.8.2.1.8) such that the activity contained in the fuel-cladding gap is released to the reactor coolant. ~~In addition, a small fraction of fuel is assumed to melt and release core inventory to the reactor coolant~~No fuel melt is calculated to occur as a result of the rod ejection (see subsection 15.4.8.2.1.8).

15.4.8.3.1 Source Term

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The ~~initial~~ reactor coolant noble gas ~~and alkali metal~~ concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fractions are modified following the guidance of Draft Guide 1199 (Reference 25), which incorporates the effects of enthalpy rise in the fuel following the reactivity insertion, consistent with Appendix B of SRP 4.2, Revision 3 (Reference 24). Draft Guide 1199 included expanded guidance for determining nuclide gap fractions available for release following a rod ejection. Reference 26 was

issued as a clarification to the gap fraction guidance in Draft Guide 1199. An enthalpy rise of 60 cal/gm is used to calculate the gap fractions (see subsection 15.4.8.2.1.8) ~~is increased to 10 percent of the inventory for iodine and noble gases and 12 percent for alkali metals~~. Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor. No fuel melt is calculated to occur as a result of the rod ejection (see subsection 15.4.8.2.1.8).

~~Even though no fuel centerline melting is expected, a conservative upper limit for fuel melting was determined to be 0.25 percent of the core based on the following assumptions:~~

~~1. No more than 50 percent of the rods experiencing clad damage will experience centerline melting. (Based on 10 percent of rods failing, this is 5 percent of the core.)~~

~~2. Due to the power distribution within the core, no more than 50 percent of the axial length of the affected fuel rods will experience melting. (This reduces the equivalent number of rods experiencing melting to 2.5 percent of the core.)~~

~~3. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the fuel volume will actually melt. (Based on 2.5 percent of the rods experiencing melting, the resulting fraction of the core experiencing melting is 0.25 percent.)~~

~~All of the noble gases and half of the iodines and alkali metals are assumed to be released from the melted fuel.~~

~~The initial secondary coolant activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.~~

15.4.8.3.5 Identification of Conservatisms

- The reactor coolant activities are based on conservative assumptions (refer to Table 15.4-4)~~an assumed fuel defect level of 0.25 percent~~; whereas, the activities based on the expected fuel defect level ~~is~~are far less ~~than this~~ (see Section 11.1).

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be ~~less than 1.8~~4.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and ~~less than 2.~~5.9 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem ~~total effective dose equivalent~~TEDE identified in 10 CFR Part 50.34. The phrase "well within" is taken as being 25 percent or less.

15.4.10 References

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Table 15.4-1 (Sheet 2 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
RCCA ejection accident		
1. ~~Beginning of cycle, full power~~PCMI limiting event	Initiation of rod ejection	0.00
	~~Power range high neutron flux (high setting) setpoint reached~~	~~0.03~~
	Peak nuclear power occurs	0.14
	~~Rods begin to fall into core~~Reactor trip setpoint reached	<0.~~9~~30
	Peak cladding temperature occurs	0~~2~~.36
	Peak ~~heat flux~~enthalpy deposition occurs	~~2.37~~0.44
	Rods begin to fall into core~~Peak fuel center temperature occurs~~	~~4.54~~1.20
2. ~~Beginning of cycle, zero power~~Peak cladding temperature limiting event	Initiation of rod ejection	0.00
	Power range high neutron flux (low setting) setpoint reached	0.37
	Peak nuclear power occurs	~~0.44~~0.08
	~~Rods begin to fall into core~~Minimum DNBR occurs	0.11~~.27~~
	Peak cladding temperature occurs ~~Peak heat flux occurs~~	0.11~~.53~~
	Reactor trip setpoint reached~~Peak cladding temperature occurs~~	<~~2.55~~0.30
	Rods begin to fall into core~~Peak fuel center temperature occurs~~	~~3.3~~1.20
3. ~~End of cycle, full power~~Peak enthalpy/peak fuel centerline temperature event	Initiation of rod ejection	0.00
	~~Power range high neutron flux (high setting) setpoint reached~~	~~0.035~~
	Peak nuclear power occurs	0.~~14~~06
	~~Rods begin to fall into core~~Reactor trip setpoint reached	<0.~~94~~30
	Rods begin to fall into core ~~Peak cladding temperature occurs~~	1.20~~.36~~
	Peak fuel center temperature occurs~~Peak heat flux occurs~~	2.~~37~~50
	Peak cladding temperature occurs~~Peak fuel center temperature occurs~~	~~4.34~~2.80

Table 15.4-1 (Sheet 3 of 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
~~4. End of cycle, zero power~~	~~Initiation of rod ejection~~	~~0.00~~
	~~Power range high neutron flux (low setting) setpoint reached~~	~~0.23~~
	~~Peak nuclear power occurs~~	~~0.27~~
	~~Rods begin to fall into core~~	~~1.13~~
	~~Peak cladding temperature occurs~~	~~1.83~~
	~~Peak heat flux occurs~~	~~1.85~~
	~~Peak fuel center temperature occurs~~	~~2.94~~

~~Table 15.4-3~~

~~PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT~~

~~Time in Life~~	~~HZP(1) Beginning~~	~~HFP(2) Beginning~~	~~HZP End~~	~~HFP End~~
~~Power level (%)~~	~~0~~	~~102(3)~~	~~0~~	~~102(3)~~
~~Ejected rod worth (%Δk)~~	~~0.65~~	~~0.37~~	~~0.75~~	~~0.30~~
~~Delayed neutron fraction (%)~~	~~0.49~~	~~0.49~~	~~0.44~~	~~0.44~~
~~Feedback reactivity weighting~~	~~2.155~~	~~1.22~~	~~2.9~~	~~1.35~~
~~Trip reactivity (%Δk)~~	~~2.0~~	~~4.0~~	~~2.0~~	~~4.0~~
~~F_q before rod ejection~~	~~–~~	~~2.6~~	~~–~~	~~2.6~~
~~F_q after rod ejection~~	~~12.0~~	~~4.9~~	~~19.6~~	~~6.0~~
~~Number of operational pumps~~	~~2~~	~~4~~	~~2~~	~~4~~
~~Maximum fuel pellet average temperature (°F)~~	~~2573~~	~~4118~~	~~2848~~	~~3926~~
~~Maximum fuel center temperature (°F)~~	~~3018~~	~~4974~~	~~3263~~	~~4871~~
~~Maximum cladding average temperature (°F)~~	~~1907~~	~~2265~~	~~2122~~	~~2151~~
~~Maximum fuel stored energy (cal/g)~~	~~104~~	~~181~~	~~117~~	~~170~~
~~Percent of fuel melted at hot spot~~	~~0~~	~~<10~~	~~0~~	~~<10~~

~~**Notes:**~~

1. ~~HZP – Hot zero power~~
2. ~~HFP – Hot full power~~
3. ~~The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.~~Table 15.4-3 Not Used

Table 15.4-4 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g (2.22E+06 Bq/g) of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g (1.036E+07 Bq/g) dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1~~0~~% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in ~~failed/melte~~damaged fuel)	1.7~~6~~5
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fuel enthalpy increase (cal/gm)	60
Fission product gap fractions	
Iodine~~s~~ 131	0.1238
Iodine 132	0.1338
Krypton 85	0.5120
~~and~~Other noble gases	0.1238
Other halogens	0.0938
Alkali metals	~~0.12~~0.6860
~~Core melting~~	
~~Fraction of core melting~~	~~0.0025~~
~~Fraction of activity released~~	
~~Iodines and alkali metals~~	~~0.5~~
~~Noble gases~~	~~1.0~~
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3 ~~in Appendix 15A~~
Nuclide data	See Table 15A-4 ~~in Appendix 15A~~

Reactor coolant mass (lb)	3.7 E+05 (1.68E+05 kg)

Note:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

Table 15.4-4 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

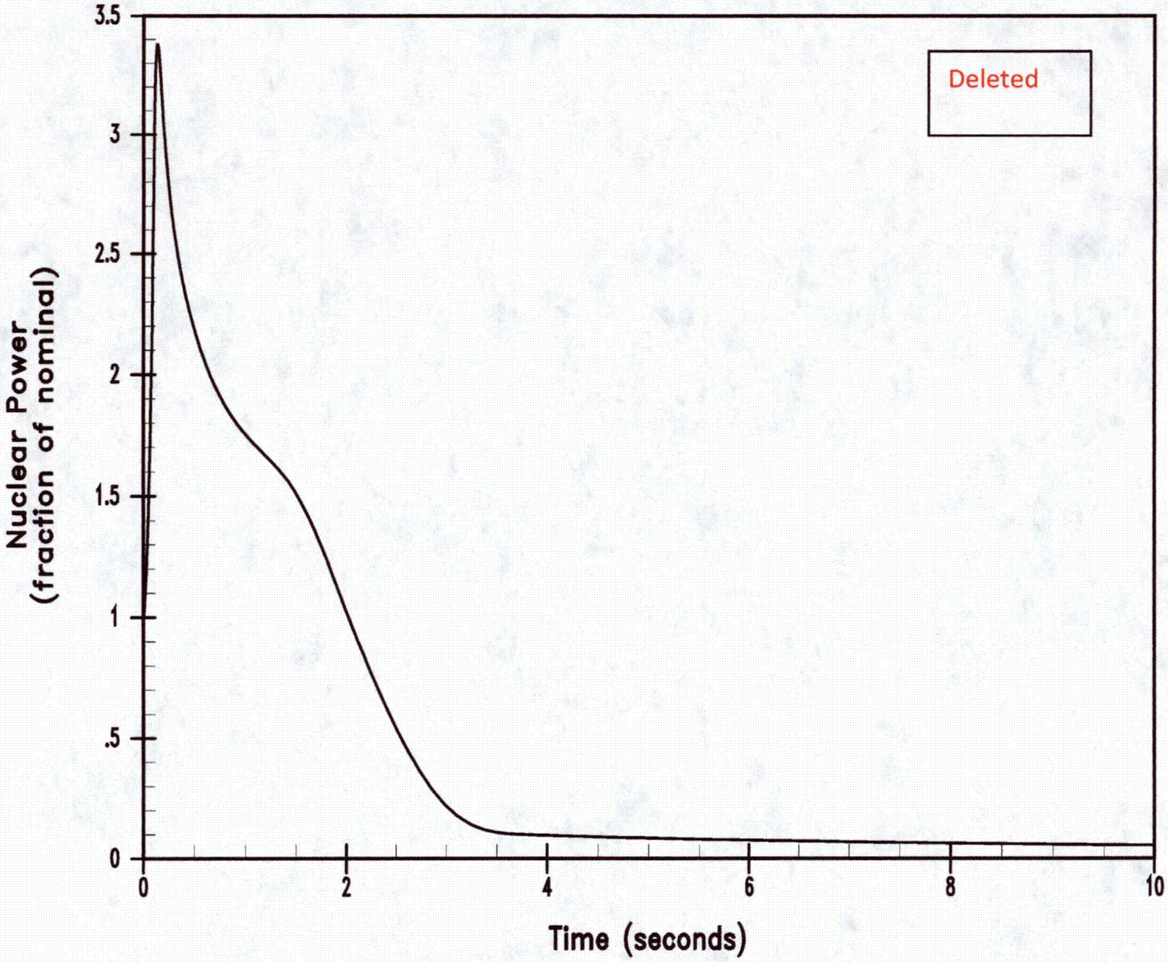
Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 ~~in Appendix 15A~~
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.~~3~~5[a] (47.4 kg/hr)
– Leak flashing fraction	0.04[b]
– Secondary coolant mass (lb)	6.06 E+05 (2.75E+05 kg)
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05 (4.90E+04 kg)
– Partition coefficient in steam generators	
• Iodine	0.01
• Alkali metals	0.0035~~1~~
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr	0.10
• >24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine	1.~~7~~9[c]
• Organic iodine	0
• Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	~~3.1~~2.78

Notes:

a. Equivalent to 300 gpd (1.14 m^3/day) cooled liquid at 62.4 lb/ft^3 (999.6 kg/m^3).

b. No credit for iodine partitioning is taken for flashed leakage.

c. From Appendix 15B.


Deleted
Nuclear Power
(fraction of nominal)
3.5
3
2.5
2
1.5
1
.5
0
0
2
4
6
8
10
Time (seconds)

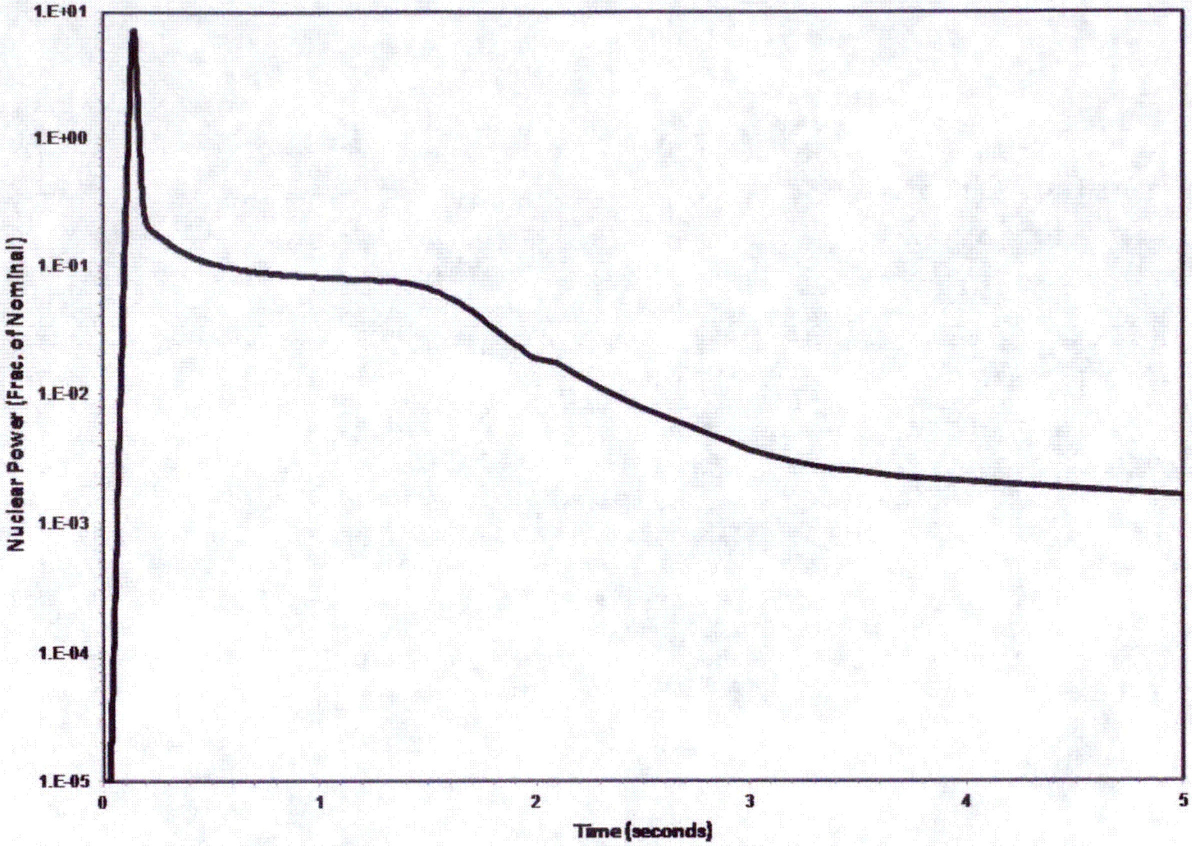
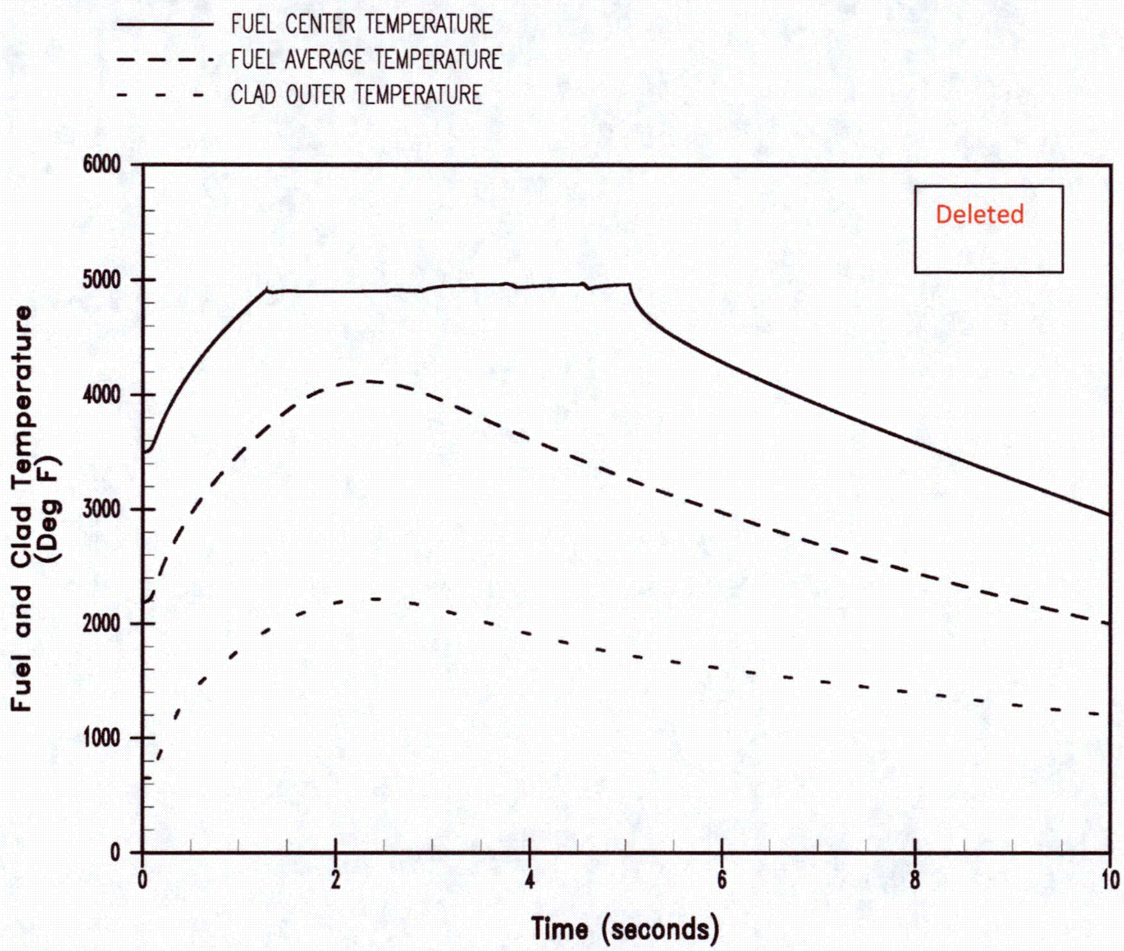


Figure 15.4.8-1

Nuclear Power Transient Versus Time
for the PCMI Rod Ejection Accident ~~at Beginning of Life, Full Power~~


FUEL CENTER TEMPERATURE
FUEL AVERAGE TEMPERATURE
CLAD OUTER TEMPERATURE
Deleted
Fuel and Clad Temperature (Deg F)
Time (seconds)
0
1000
2000
3000
4000
5000
6000
0
2
4
6
8
10

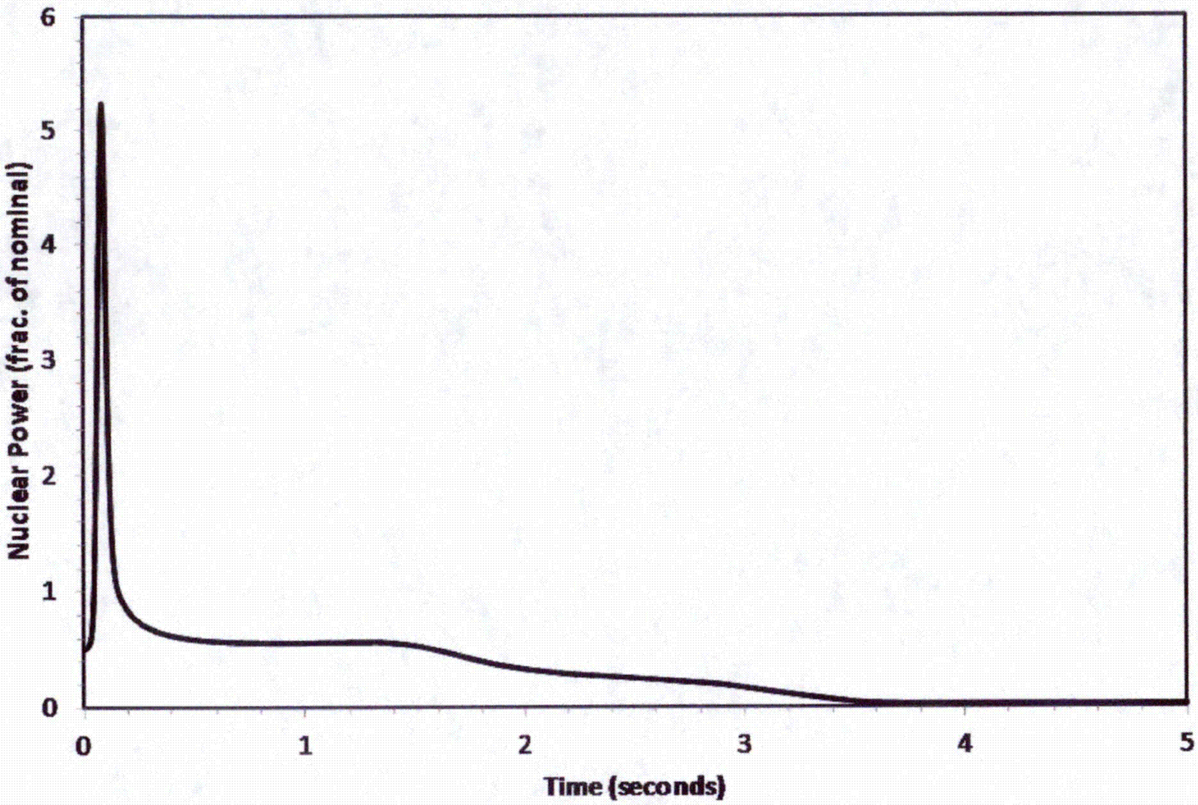
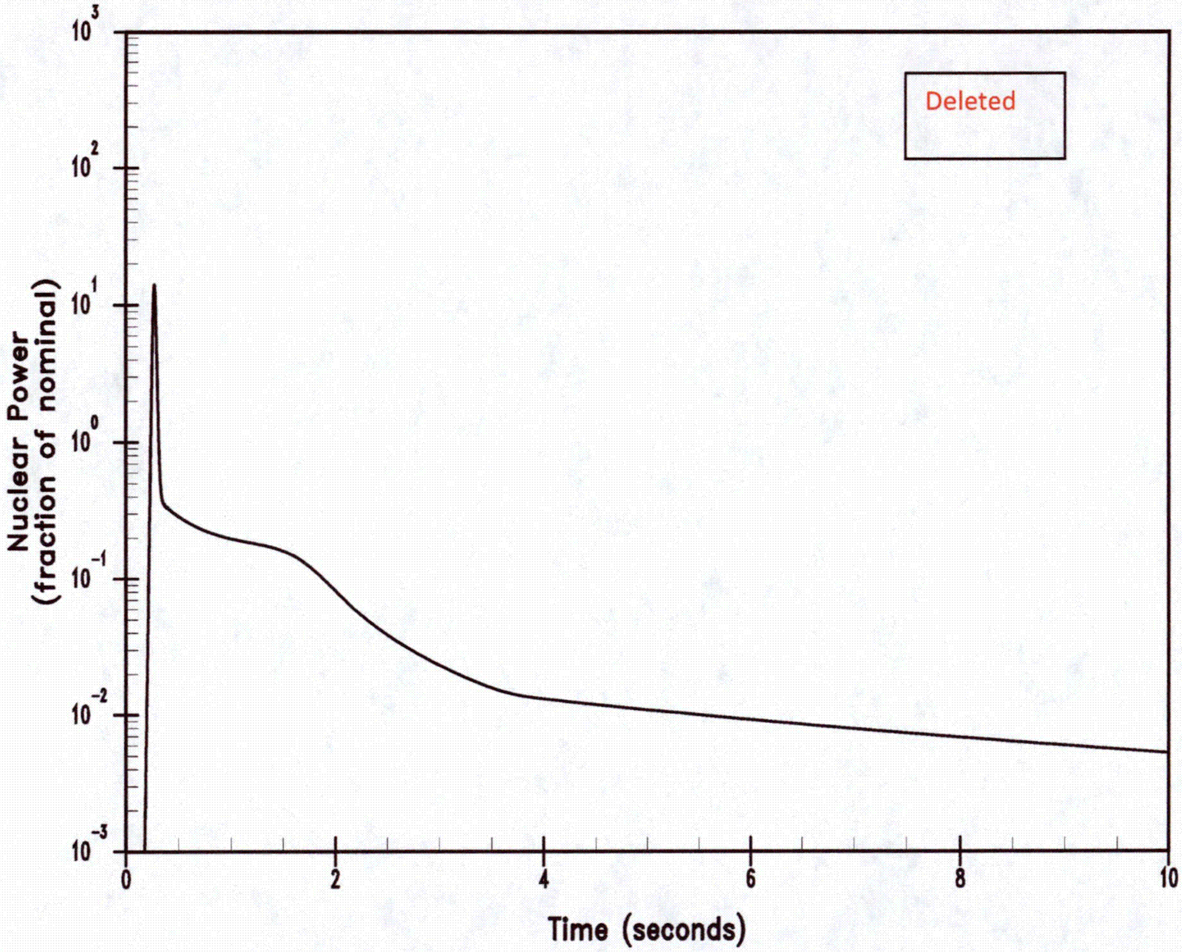


Figure 15.4.8-2

Nuclear Power Transient ~~Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature~~ Versus Time ~~at Beginning of Life, Full Power~~for the High Cladding Temperature Rod Ejection Accident


Deleted
Nuclear Power
(fraction of nominal)
10^3
10^2
10^1
10^0
10^{-1}
10^{-2}
10^{-3}
0
2
4
6
8
10
Time (seconds)

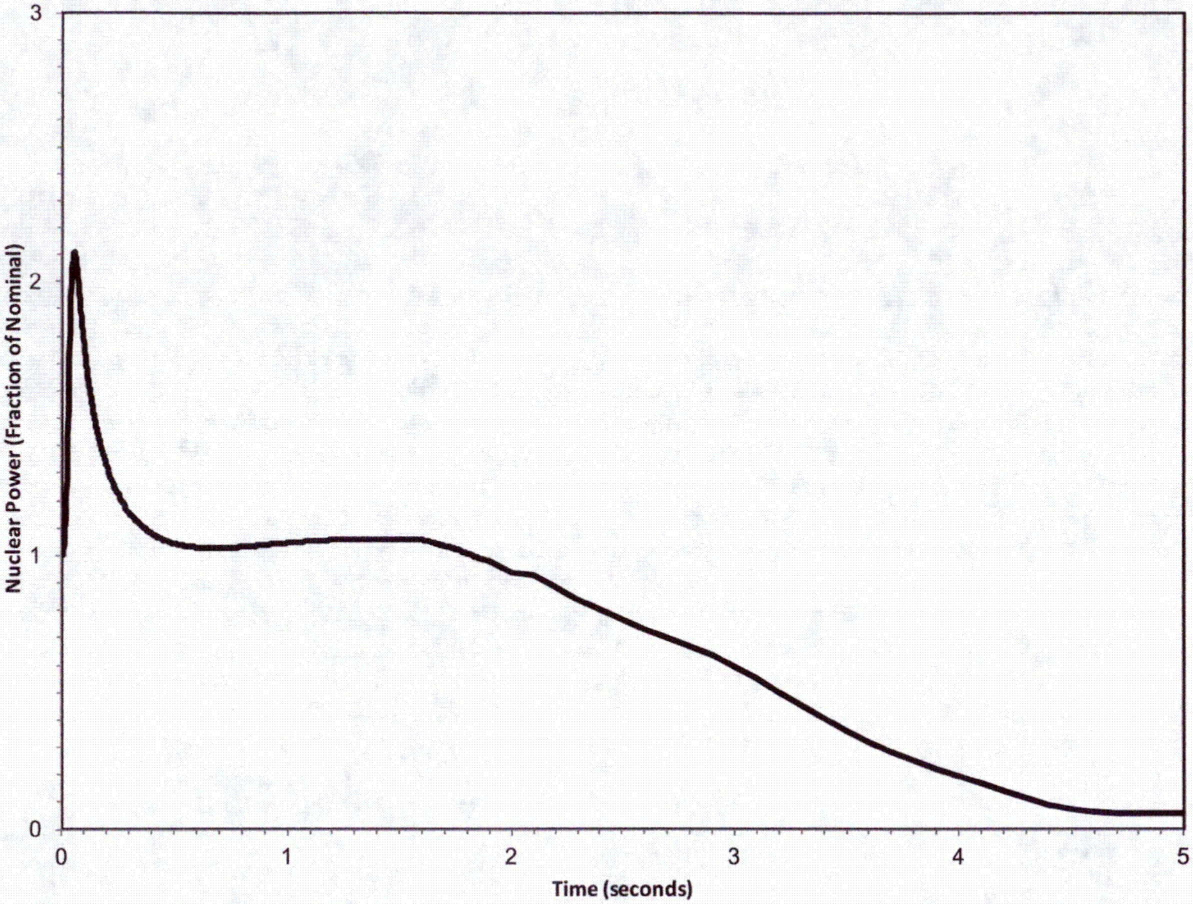
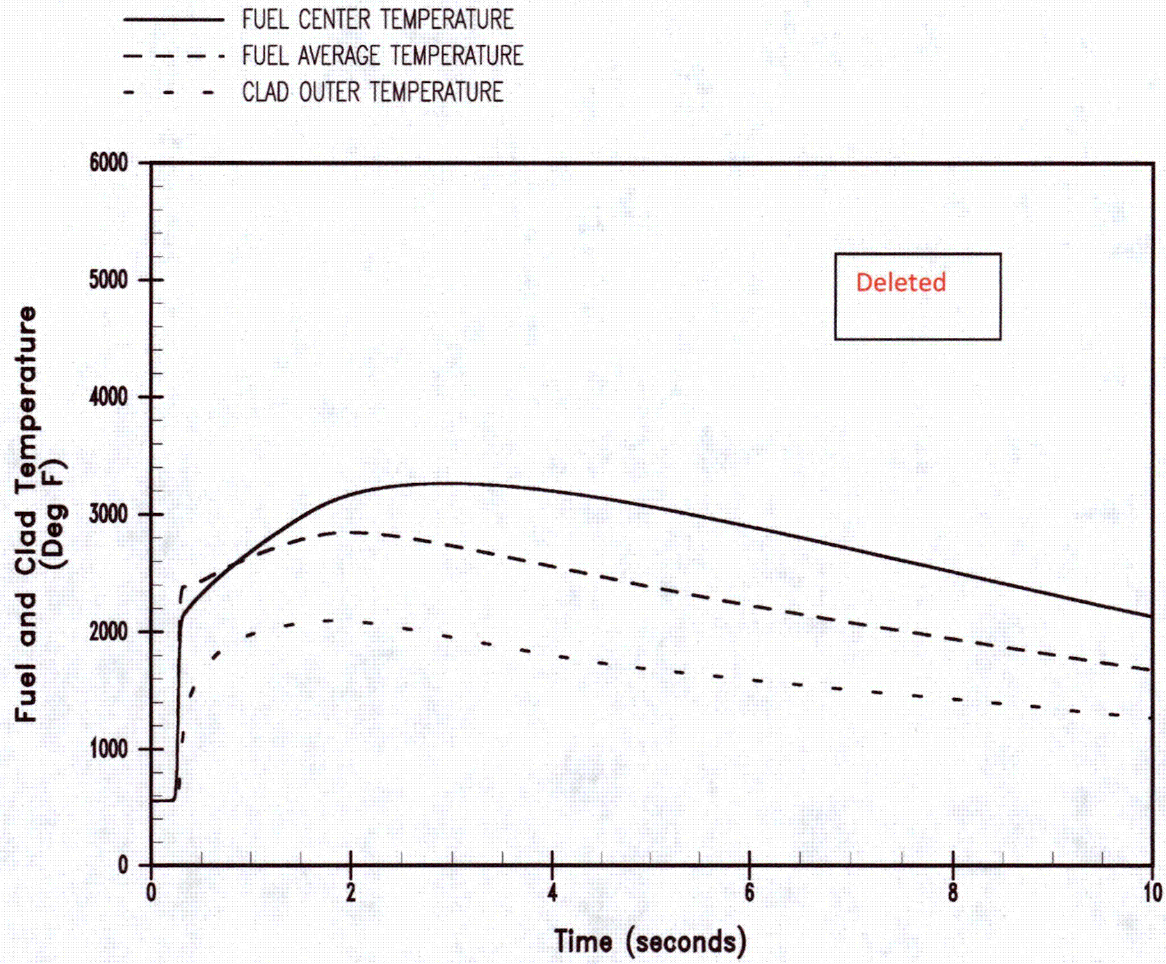


Figure 15.4.8-3

Nuclear Power Transient Versus Time ~~at End of Life, Zero Power~~for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident

Figure 15.4.8-4 not used



~~Figure 15.4.8-4~~

~~Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at End of Life, Zero Power~~

15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be ≤ 1.3~~1~~ rem at the exclusion area boundary and ≤ 0.~~5~~6 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "a small fraction" is taken as being ten percent or less.

15.6.3.3.1 Source Term

The secondary coolant iodine and alkali metal activity is assumed to be 1~~0~~ percent of the maximum equilibrium primary coolant activity.

15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be ~~less than~~ 0.7~~6~~ rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) and ~~less than~~ 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be ~~less than~~ 1.4 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and ~~less than~~ 0.7 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

Table 15.6.2-1

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SMALL LINE BREAK OUTSIDE CONTAINMENT

Reactor coolant iodine activity	Initial activity equal to the design basis reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Table 15A-2 in Appendix 15A)[(a)]
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Break flow rate (gpm)	130[(b)]
Fraction of reactor coolant flashing	0.47~~1~~
Duration of accident (hr)	0.5
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Nuclide data	See Table 15A-4

Notes:

a. Use of accident-initiated iodine spike is consistent with the guidance in the Standard Review Plan.

b. At density of 62.4 lb/ft^3.

Table 15.6.3-3 **PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE**	
Reactor coolant iodine activity – Accident initiated spike – Preaccident spike	 Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A). Duration of spike is ~~5.3~~8.0 hours. An assumed iodine spike that results in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal	1~~0~~% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.~~84~~7 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	~~13~~15.~~19~~94
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop – Initial secondary coolant mass (lb) – Primary-to-secondary break flow – Integrated flashed break flow (lb) – Steam released (lb) – Iodine partition coefficient – Alkali metals partition coefficient	 1.1~~6~~6 E+05 See Figure 15.6.3-5 See Figure 15.6.3-10 See Table 15.6.3-2 1.0 E-02[(a)] ~~1~~3.~~0~~5 E-03[(a)]
Steam generator in intact loop – Initial secondary coolant mass (lb) – Primary-to-secondary leak rate (lb/hr) – Steam released (lb) – Iodine partition coefficient – Alkali metals partition coefficient	 2.~~00~~30 E+~~05~~04 52.~~14~~16[(b)] See Table 15.6.3-2 1.0 E-02[(a)] ~~1~~3.~~0~~5 E-03[(a)]

Notes:

a. Iodine ~~P~~partition coefficient does not apply to flashed break flow.

b. Equivalent to 150 gpd at psia cooled liquid at 62.4 lb/ft^3.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when ~~high~~ High-2 iodine or particulate activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. The bottled air also induces flow through the passive air filtration system which filters contaminated air in the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply. As a defense-in-depth measure, the nonsafety-related normal control room HVAC would be brought back into operation with the supplemental filtration train if power is available.

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The design of the emergency habitability system (VES) provides 65 scfm ±5 scfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this should result in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. However, no additional credit is taken for dilution of the vestibule via the purge. The projected inleakage into the main control room through ingress/egress is 5 cfm. An additional 10 cfm of unfiltered inleakage is conservatively assumed from other sources.

Activity entering the main control room is assumed to be uniformly dispersed. With the VES in operation, airborne activity is removed from the main control room atmosphere via the passive recirculation filtration portion of the VES.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.8.1 Offsite Doses

The reported exclusion area boundary doses are for the time period of ~~1.4~~1.3 to ~~3.4~~3.3 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings, shine from radioactivity accumulated on the VES or VBS filters, and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of ~~the three~~these dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and is reported in Table 15.6.5-3.

Table 15.6.5-2 (Sheet 1 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Primary coolant source data	
– Noble gas concentration	280 µCi/g dose equivalent Xe-133
– Iodine concentration	1.0 µCi/g dose equivalent I-131
– Primary coolant mass (lb)	~~3.72~~4.39 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	~~880~~16000
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	~~0.5~~1.0
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Elemental iodine deposition removal coefficient (hr^{-1})	~~1.7~~1.9
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room volume (ft^3)	~~35,700~~3.89E+04
– Volume of HVAC, including main control room and control support area (ft^3)	~~105,500~~1.2E+05
– Normal HVAC operation (prior to switchover to an emergency mode)	
• Air intake flow (cfm)	~~1925~~1650
• Filter efficiency	Not applicable
– Atmospheric dispersion factors (sec/m^3)	See Table 15A-6

Table 15.6.5-2 (Sheet 2 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Main control room model (cont.)	
– Occupancy	
• 0 - 24 hr	1.0
• 24 - 96 hr	0.6
• 96 - 720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04
Control room with emergency habitability system credited (VES Credited)	
– Main control room activity level at which the emergency habitability system actuation is actuated (Ci/m^3 of dose equivalent I-131)	~~2.0 E-~~2.0 E-~~06~~07
– Response time to actuate VES based on radiation monitor response time and VBS isolation (sec)	~~180~~200
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Unfiltered inleakage via ingress/egress (scfm)	5
• Unfiltered inleakage from other sources (scfm)	10
• Recirculation flow through filters (scfm)	600
• Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	~~30~~90
• Particulates	99
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	~~1700~~1900
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply is restored and emergency habitability system returns to operation (hr)	168

Table 15.6.5-2 (Sheet 3 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Control room with credit for continued operation of HVAC (VBS Supplemental Filtration Mode Credited)	
– Time to switch from normal operation to the supplemental air filtration mode (sec)	~~60~~265
– Unfiltered air inleakage (cfm)	25
– Filtered air intake flow (cfm)	860
– Filtered air recirculation flow (cfm)	2740
– Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
Miscellaneous assumptions and parameters	
– Offsite power	Not applicable
– Atmospheric dispersion factors (offsite)	See Table 15A-5
– Nuclide dose conversion factors	See Table 15A-4
– Nuclide decay constants	See Table 15A-4
– Offsite breathing rate (m^3/sec)	
0 - 8 hr	3.5 E-04
8 - 24 hr	1.8 E-04
24 - 720 hr	2.3 E-04

Table 15.6.5-3

RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT WITH CORE MELT

	TEDE Dose (rem)
Exclusion zone boundary dose (~~1.4~~1.3 - ~~3.4~~3.3 hr)[(1)]	~~24.6~~23.5
Low population zone boundary dose (0 - 30 days)	~~23.4~~22.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	~~4.25~~3.70
– Direct radiation from adjacent structures, including sky shine	~~0~~0.~~.15~~30
– Filter shine	0.32 ~~0.01~~
– Spent fuel pooling boiling	0.01
– Total	~~4.~~4.~~41~~33
Main control room dose (normal HVAC operating in the supplemental filtration mode)	
– Airborne activity entering the main control room	~~4.~~4.~~56~~50
– Direct radiation from adjacent structures, including sky shine	~~0.~~0.~~15~~30
– Filter shine	~~0.~~0.~~01~~03
– Spent fuel pooling boiling	0.01
– Total	~~4.~~4.~~73~~84

15.7.4.5 Offsite Doses

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.~~7~~ 8 rem TEDE at the site boundary and 1.2 rem TEDE at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "well within" is taken as meaning 25 percent or less.

Table 15.7-1

ASSUMPTIONS USED TO DETERMINE FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

Source term assumptions	
– Core power (MWt)	3434~~68~~[1]
– Decay time (hr)	48
Core source term after 48 hours decay (Ci)	
I-130	1.28~~2.49~~ E+05
I-131	8.18~~26~~ E+07
I-132	9.10~~27~~ E+07
I-133	4.06~~11~~ E+07
I-135	1.17~~21~~ E+06
Kr-85m	1.52~~9~~ E+04
Kr-85	1.07~~5~~ E+06
Kr-88	5.45~~81~~ E+02
Xe-131m	1.02~~5~~ E+06
Xe-133m	4.47~~37~~ E+06
Xe-133	1.70~~69~~ E+08
Xe-135m	1.~~94~~91 E+05
Xe-135	1.04~~8~~ E+07
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.7~~6~~5
Percentage of fission products in gap	
I-131	8
Other iodines	5
Kr-85	10
Other noble gases	5
Pool decontamination factor for iodine	200
Activity release period (hr)	2
Atmospheric dispersion factors	See Table 15A-5 in Appendix 15A
Breathing rates (m^3/sec)	3.5 E-4
Nuclide data	See Appendix 15A

Note:

1. The main feedwater flow measurement supports a 1-percent power uncertainty~~; use of a 2-percent power uncertainty is conservative~~.

15A.3.1.2 Secondary Coolant Source Term

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be ~~10~~ 1 percent of the primary coolant equilibrium source term. This is more conservative than using the design basis secondary coolant source terms listed in Table 11.1-5.

Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

15B.1 Elemental Iodine Removal

Elemental iodine is removed by deposition onto the structural surfaces inside the containment. The removal of elemental iodine is modeled using the equation from the Standard Review Plan (Reference 1):

$$\lambda_d = \frac{K_W A}{V}$$

where:

λ_d	=	first order removal coefficient by surface deposition
K_w	=	mass transfer coefficient (specified in Reference 1 as 4.9 m/hr)
A	=	surface area available for deposition
V	=	containment building volume

The available deposition surface is ~~219,000~~251,000 ft^2, and the containment building net free volume is 2.06×10^6 ft^3. From these inputs, the elemental iodine removal coefficient is ~~1.7~~1.9 hr^{-1}.

Consistent with the guidance of Reference 1, credit for elemental iodine removal is assumed to continue until a decontamination factor (DF) of 200 is reached in the containment atmosphere. Because the source term for the LOCA (defined in subsection 15.6.5.3) is modeled as a gradual release of activity into the containment, the determination of the time at which the DF of 200 is reached needs to be based on the amount of elemental iodine that enters the containment atmosphere over the duration of core activity release.

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ ~~0.10~~0.01 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant ≤ ~~0.10~~0.01 μCi/gm DOSE EQUIVALENT I-131.	31 days

APPLICABLE SAFETY ANALYSES (continued)

assumed to be the LCO of 280 μCi/gm DOSE EQUIVALENT XE-133. The safety analysis assumes the specific activity of the secondary coolant at its limit of ~~0.1~~0.01 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.4, "Secondary Specific Activity."

The LCO limits ensure that, in either case, the doses reported in Chapter 15 remain bounding.

The RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube LEAKAGE from the Reactor Coolant System (RCS). Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant. While fission products present in the primary coolant, as well as activated corrosion products, enter the secondary coolant system due to the primary to secondary LEAKAGE, only the iodines are of a significant concern relative to airborne release of activity in the event of an accident or abnormal occurrence (radioactive noble gases that enter the secondary side are not retained in the coolant but are released to the environment via the condenser air removal system throughout normal operation).

The limit on secondary coolant radioactive iodines minimizes releases to the environment due to anticipated operational occurrences or postulated accidents.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (SLB) as discussed in Chapter 15 (Ref. 1) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~0.1~~0.01 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a postulated SLB are within the acceptance criteria in SRP Section 15.0.1, and within the exposure guideline values of 10 CFR Part 50.34.

Secondary specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit of the secondary coolant is required to be ≤ ~~0.1~~0.01 μCi/gm DOSE EQUIVALENT I-131 to maintain the validity of the analyses reported in Chapter 15 (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System (VES)

BASES

BACKGROUND

The Main Control Room Emergency Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 iodine or particulate Main Control Room Envelope (MCRE) radiation signal is received, the VES is actuated. The MCRE radioactivity is measured by detectors in the MCR supply air duct, downstream of the filtration units. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air (Ref. 4) for the MCRE occupants; 2) to provide forced ventilation to maintain the MCRE at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; 3) provide passive filtration to filter contaminated air in the MCRE; and 4) to limit the temperature increase of the MCRE equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

The VES consists of compressed air storage tanks, two air delivery flow paths, an eductor, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), associated valves or dampers, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCRE for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCRE occupants.

The MCRE is the area within the confines of the MCRE boundary that contains the spaces that control room operators inhabit to control the unit during normal and accident conditions. This area encompasses the main control area, operations work area, operational break room, shift supervisor's office, kitchen, and toilet facilities (Ref. 1). The MCRE is protected during normal operation, natural events, and accident conditions. The MCRE boundary is the combination of walls, floor, roof, electrical and mechanical penetrations, and access doors. The OPERABILITY of the MCRE boundary must be maintained to ensure that the inleakage of unfiltered air into the MCRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to MCRE occupants. The MCRE and its boundary are defined in the Main Control Room Envelope Habitability Program.

BACKGROUND (continued)

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCRE, which is initially at or below 75°F. Heat sources inside the MCRE include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCRE pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCRE habitability is maintained by operating one of the two MCRE ancillary fans to supply outside air to the MCRE.

The compressed air storage tanks are initially filled to contain greater than 327,574 scf of compressed air. The compressed air storage tanks, the tank pressure, and the room temperature are monitored to confirm that the required volume of breathable air is stored. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCRE. The MCRE is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas. The VES operation in maintaining the MCRE habitable is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The compressed air storage tanks are sized such that the set of tanks has a combined capacity that provides at least 72 hours of VES operation.

Operation of the VES is automatically initiated by either of the following safety related signals:

- Control Room Air Supply Iodine or Particulate Radiation - H~~h~~igh-2
- Loss of all AC power for more than 10 minutes~~particulate or iodine radioactivity~~.

~~In the event of a loss of all AC power, the VES functions to provide ventilation, pressurization, and cooling of the MCRE pressure boundary.~~

In the event ~~of~~ that a ~~h~~High-1 ~~level of gaseous~~ radioactivity setpoint value is reached ~~outside of the MCRE~~, the non-safety VBS ~~continues to operate to provide pressurization and~~ re-aligns to supplemental filtration mode, providing ~~filtration functions. The~~ MCRE pressurization, cooling, and filtration. ~~air supply downstream of the filtration units is monitored by a safety related radiation detector.~~

Upon ~~h~~High-2 particulate or iodine radioactivity setpoint, a safety related signal is generated to isolate the MCRE and to initiate air flow from the VES storage tanks. Isolation of the MCRE consists of closing safety related valves in the lines that penetrate the MCRE pressure boundary. Valves in the VBS supply and exhaust ducts, and the Sanitary Drainage System (SDS) vent lines are automatically isolated. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.