14.11 ANALYSIS OF DESIGN BASIS ACCIDENTS - PRE-UPRATED

This section does not reflect the effects from power uprate. For power uprated conditions, the results of the evaluation at 3458 MWt are provided in Section 14.6.

14.11.1 Introduction

The methods described in Subsection 14.4 for identifying and evaluating accidents have resulted in the establishment of design basis accidents for the various accident categories as follows:

Accident Category	Design Basis Accident
a. Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact.	Rod drop accident (single control rod)
b. Accidents that result in radioactive material release directly to the primary containment.	Loss-of-coolant accident (rupture of one recirculation loop).
c. Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact.	Accidents in this category are less severe than those in categories "d" and "e", below.
d. Accidents that result in radioactive material release directly to the secondary containment with the primary containment not intact.	Refueling accident (fuel assembly drops on spent fuel during refueling).
e. Accidents that result in radioactive material releases outside the secondary containment.	Steam line break accident (main steam line breaks outside of secondary containment).

An investigation of accident possibilities reveals that accidents in category "c" are less severe than those in categories "d" and "e". There are two varieties of accidents in category "c": (1) failures of the nuclear system process barrier inside the secondary containment, and (2) failures involving fuel that is located outside the primary containment but inside the secondary containment. Under the accident selection rules described in Subsection 14.4, a main steam line break inside the reactor building is the most severe accident of the first variety, but this accident results in a radioactivity release to the environs no greater than that resulting from the main steam line break outside the secondary containment. Similarly, the most severe accident of the second variety is the dropping of a fuel assembly into the fuel pool during refueling. Because the consequences of accidents in category "c" are less severe than those resulting from similar accidents in other categories, the accidents in category "c" are not described.

14.11.2 Control Rod Drop Accident (CRDA)

The accidents that result in releases of radioactive material from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact are the results of various failures of the Control Rod Drive System. Examples of such failures are collet finger failures in one control rod drive mechanism, a control drive system pressure regulator malfunction, and a control rod drive mechanism ball check valve failure. None of the single failures associated with the control rods or the control rod system results in a greater release of radioactive material from the fuel than the release that results when a single control rod drops out of the core after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position. Thus, this control rod drop accident is established as the design basis accident for the category of accidents resulting in radioactive material release from the fuel with all other barriers initially intact. A highly improbable combination of actual events would be required for the design basis control rod drop accident to occur. The actual events required are as follows:

a. Failure of the rod-to-drive coupling. The design of the coupling itself reduces the probability of separation. Tests conducted under both simulated reactor conditions and the conditions more extreme than those expected in reactor service have shown that the coupling does not separate, even after thousands of scram cycles. Tests also show that the coupling does not separate when subjected to forces 30 times greater than that which can be achieved by normal control rod drive operation.

b. Sticking of the control rod in its fully inserted position as the drive is withdrawn. The control rods are designed to minimize the probability of sticking in the core. The control rod blades, which are equipped with rollers or spacer pads at the top of the control rod blade and rollers at the bottom that make contact with the channel walls, travel in gaps between the fuel assemblies with approximately 1/2-inch total clearance. Control rods of similar design, now in use in operating reactors, have exhibited no tendency to stick in the core due to distortion or swelling of the blade.

c. Full withdrawal of the control rod drive.

d. Failure of the operator to notice the lack of response of neutron monitoring channels as the rod drive is withdrawn.

e. Failure of the operator to verify rod coupling. The control rod bottoms on a seal, preventing the control rod drive from being withdrawn at the overtravel position. Attempting to withdraw a control rod drive to the overtravel position provides a method for verifying rod coupling: this verification is required whenever neutron monitoring equipment response does not indicate that the rod is following the drive.

The accident is analyzed over the full spectrum of power conditions. Nuclear excursion results are presented for three points in this range: the cold (68°F) critical condition for moderator and fuel, a hot (547°F) critical condition, and the 10 percent of rated power condition. The results of the rod drop accident initiated from higher than 10 percent power are less severe than the 10 percent power case because of the faster doppler response. Only the radiological results of the most severe case are presented.

Subsections 14.11.2.1 through 14.11.2.7 discuss the CRDA and the analysis performed for the initial core loading. It is retained in the FSAR because the information presented is useful to understanding the Control Rod Drop accident and the related licensing bases for the initial core and cycle. A complete and detailed discussion of the CRDA including accident description, causes, sequence of events, consequences of the accident, and the accident analysis (analysis methods, assumptions, conditions, results and consequences) using refined analytical methods is given in the licensing topical report, "GESTAR II," NEDE - 24011-P-A, May 1986 and revisions thereto. Subsection 14.11.2.8 discusses the current CRDA performed for BFN. This analysis documents the safety design basis for eliminating the Main Steam Line Radiation Monitors as required components to mitigate a CRDA.

The CRDA is a limiting event that is impacted by core and fuel design and thus it must be considered for each reload cycle. An improved Rod Worth Minimizer incorporating a "Banked Position Withdrawal Sequence" (BPWS) has been developed which greatly reduces the maximum control rod worth that could occur during an CRDA such that in all cases the peak fuel enthalpy is much less than the acceptance criteria of 280 cal/gm. For reload cycles in which the BPWS is utilized a cycle specific CRDA analysis is not required. The cycle specific CRDA results or a commitment to employ BPWS are contained in the Reload Licensing Report.

14.11.2.1 Initial Conditions

The following initial conditions were assumed for the three cases presented in the initial CRDA analysis:

Case A (cold):	Reactor critical Moderator and fuel at 68°F Power level 10^{-8} x design Rod worth (for dropped rod) 0.025 Δk.
Case B (hot):	Reactor critical Moderator and fuel at 547°F Power level 10-6 x design Rod worth (for dropped rod) 0.025 Δk.
Case C (power)	Reactor critical Moderator and fuel at 547°F Power level 10-1 x design Rod worth (for dropped rod) 0.038 Δk.

In considering the possibilities of a control rod drop accident, only the rod worths of the lower curve of Figure 14.11-1 are pertinent at less than ten percent power. These are the rods which are normally allowed to be moved by operating procedures and the rod worth minimizer. The non scheduled rods, those described by the central envelope, do not have a withdrawal permissive during the time their worths are greater than the lower curve, so they are held full in by the control rod drive and cannot drop from the core. If a nonscheduled rod were selected, the rod worth minimizer blocks rod movement. Therefore, the worth of the strongest rod which could be stuck is limited to about 0.01 Δk, and the 0.025 Δk worth assumed for cases A and B is considerably above the rod worth values available for stuck rods under the assumed reactor conditions. In the greater than ten percent power range, the maximum rod worth is determined by the FLARE[1] and WANDA[2] computer codes and is shown in Figure 14.11-2. Thus, in case C the rod worth is assumed to be 0.038 Δk.

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[1] Delp, D. L., et al.: "FLARE-A Three Dimensional Boiling Water Reactor Simulator," GEAP-4598, July 1964.

[2] Marlowe, O. C., and Suggs, M. C.: "WANDA-5-A One Dimensional Neutron Diffusion Equation Program for the Philco 2000 Computer," WAPD-TM-241, November, 1960.

14.11.2.2. Excursion Analysis Assumptions

The following assumptions are used in the analysis of the nuclear excursion for each case:

a. The velocity at which the control rod falls out of the core is assumed to be 5 ft/sec. The control rod velocity limiter[3] an engineered safeguard, limits the rod drop velocity to less than this value.

b. Control rod scram motion is assumed to start at about 200 milliseconds after the neutron flux has attained 120 percent of rated flux. This assumption allows the power transient to be terminated initially by the Doppler reactivity effect of the fuel. This assumption is particularly conservative for cases A and B because a high neutron flux scram would be initiated earlier by the intermediate range neutron monitoring channels (IRM).

c. No credit is taken for the negative reactivity effect resulting from the increased temperature of, or void formation in the moderator because the time constant for heat transfer between the fuel and the moderator is long compared with the time required for control rod motion.

d. No credit is taken for the prompt negative reactivity effect of heating in the moderator due to gamma heating and neutron thermalization.

14.11.2.3 Fuel Damage

Fuel rod damage estimates (initial core) were based upon the UO_2 vapor pressure data of Ackerman[4] and interpretation of all the available SPERT, TREAT, KIWI, and PULSTAR test results which show that the immediate fuel rod rupture threshold is about 425 cal/gm. Two especially applicable sets of data come from the PULSTAR[5] and ANL-TREAT[6] tests. The PULSTAR tests, which used UO_2 pellets of six percent enrichment with Zr-2 cladding, achieved maximum fuel enthalpies of about 200 cal/gm with a minimum period of 2.83 msec. The coolant flow was by natural convection. Film boiling occurred and there were local clad bulges; however, fuel pin integrity was maintained and there were no abnormal pressure rises.

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3 "Control Rod Velocity Limiter," General Electric Company, Atomic Power Equipment Department, March 1967 (APED-5446).

4 Ackerman, R. J., Gilles, W. P., and Thorn, R. J.: "High Temperature Vapor Pressure of UO_2," Journal of Chemical Physics, December 1956, Vol. 25, No. 6.

5 MacPhee, J., and Lumb, R. F.: "Summary Report, PULSTAR Pulse Tests-II," WNY-020, February 1965.

6 Baker, L., Jr., and Tevebaugh, A. D.: "Chemical Engineering Division Report, January-June 1964, Section V - Reactor Safety," ANL-6900.

The two ANL-TREAT tests used Zircaloy clad UO_2 pins with energy inputs of 280 and 450 calories per gram.

	Test 1	Test 2
Input Energy (cal/gm)	280	450
Final Mean Particle Diameter (mils)	60	30
Pressure Rise Rate (psi/sec	30	60

The ultimate degree of fuel fragmentation and dispersal of the two cases is not significantly different; however, the pressure rise rate in the higher energy test is increased by a factor of 20. This strongly implies that the dispersion rate in the higher energy test was significantly higher than that of the lower energy test. This leads to the logical conclusion that although a high degree of fragmentation occurs for fuel in the 200 to 300 calories per gram range, the breakup and dispersal into the water is gradual and pressure rise rates are very modest. On the other hand, for fuel above the 400 calories per gram range, the breakup and dispersal is prompt and much larger pressure rise rates are probable.

Based on the analysis of the above referenced data, it was estimated that 170 cal/gm is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/gm range and a minimum of 425 cal/gm is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

A parametric analysis was made of the rod drop accident for various starting conditions and rod worths. The results are shown in Figures 14.11-3 and 14.11-4, and the reduction in final peak fuel enthalpy with increasing initial power level is clearly shown. The cold critical case (case A) is shown as point A on Figure 14.11-3, and the hot standby critical case (case B) is shown as point B on Figure 14.11-4. Figure 14.11-5 is a conservative description of the consequences when the core is at rated temperature and the coolant is boiling. Here the ten percent of power case (case C) is represented by point C. In these cases the maximum initial enthalpy generally is not in the fuel which experiences the greatest enthalpy addition during the excursion. If a rod were dropped from a high initial enthalpy region, the results would not be as great as with one dropped from a lower enthalpy region. However, for conservatism, it is assumed that the peak enthalpy increment is added to the maximum fuel enthalpy that existed in the vicinity of the excursion center prior to the accident.

In the hot standby critical case, the power transient is calculated to have a total energy generation of 4000 MW-seconds (approximately 1.2 full power seconds). The excursion energy is calculated to be distributed in the fuel such that about 330 fuel rods have enthalpies greater than 170 cal/gm. The maximum UO_2 enthalpy is calculated to be 220 cal/gm. Essentially no fuel will melt because fuel melting occurs in the range from 220 to 280 cal/gm.

The power transient in the ten percent of power rod drop accident is less severe than the one at hot standby. The peak enthalpy is about 200 cal/gm and only about 50 fuel rods have enthalpies exceeding 170 cal/gm.

The power transient in the cold condition rod drop accident is calculated to be distributed in the fuel such that about 200 fuel rods have enthalpies greater than 170 cal/gm. The maximum UO_2 enthalpy is calculated to be 250 cal/gm. Approximately 50 pounds of UO_2 have enthalpies in excess of 220 cal/gm. Because fewer fuel rods are perforated and because the shutdown cooling system would be operating, allowing no radioactivity release to the main condenser, the radiological results of the cold rod drop accident are insignificant when compared to the hot standby critical case.

All of these peak enthalpies are far below 425 cal/gm which is estimated to be the threshold for immediate rupture of fuel rods due to UO_2 vapor pressure. Furthermore, the above peak enthalpies are well below the design limit of 280 cal/gm. Thus, there are no damaging pressure pulses as a result of the rod drop accident; and the only damage expected would be the failed fuel rods.

14.11.2.4 Fission Product Release From Fuel

The following assumptions were used in the initial calculation of fission product activity release from the fuel:

a. Three hundred thirty fuel rods fail. This is the largest number of failed fuel rods resulting from the analysis of the rod drop accident over the full spectrum of power conditions.

b. The reactor has been operating at design power until 30 minutes before accident initiation. When translated into actual plant operations, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 minutes of the departure from design power. The 30-minute time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory for the calculation.

c. The reactor has been operating at design power for 1,000 days prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. Longer operating histories do not increase the concentration of longer lived fission products significantly.

d. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity in a perforated fuel rod are assumed released. These release percentages are consistent with actual measurements made on defective fuel experiments. The basis for these values is presented in APED-57561.[7]

e. The following fission product concentrations in all fuel rod plenums are applicable for the core at the time the accident occurs:

Noble gases	4.5×10^8 Ci
Halogens	8.3×10^8 Ci

These concentrations are the result of a nuclear analysis of the fuel assuming operation at design power for 1,000 days followed by a 30-minute decay period.

f. None of the solid fission products is released from the fuel. Because the fraction of solid fission product activity available for release from the fuel is negligible, this assumption is reasonable.

g. The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible in comparison with the concentration of fission products already assumed present in the fuel.

Using the above assumptions, the following amounts of fission product activity are released from the failed fuel rods to the reactor coolant:

Noble gases (Xe,Kr)	7.1×10^4 Ci
Halogens (Br,I)	2.4×10^4 Ci

14.11.2.5 Fission Product Transport

The following assumptions were used in calculating the amounts of fission product activity transported from the reactor vessel to the main condenser (initial core):

a. The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is five percent of rated. The 25 percent recirculation flow and five percent steam flow are the maximum flow rates expected when the reactor is being taken to power and the main condenser is still being evacuated by the

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7 Horton, N. R., Williams, W. A., And Holtzclaw, J. W. "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," General Electric Company, Atomic Power Equipment Department, March 1969, (APED-5756).

mechanical vacuum pump. The recirculation flow rate is used in determining the volume of coolant in which the activity released from the fuel is deposited. The five percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would actually be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steam isolation valves to achieve full closure.

b. The main steam isolation valves are assumed to receive an automatic closure signal 0.5 seconds after the radiation monitors are tripped and to be fully closed at 10 seconds from the receipt of the closure signal. The automatic closure signal originates from the main steam line radiation monitors. The 10-second closure time of the main steam isolation valves is the maximum closing time permitted by valve setting. The total time required to isolate the main steam lines (10.5 seconds), combined with the assumptions in "a", allows calculation of the total amount of fission product activity transported to the condenser before the steam lines are isolated.

c. All of the noble gas activity is assumed released to the steam space of the reactor vessel. None is retained in the liquid reactor coolant.

d. The ratio of the halogen concentration in steam to that of water is assumed to be 3×10^{-5} by volume. Measurements, taken under applicable chemical and physical conditions at Dresden Nuclear Power Station Unit No. 1, indicate that the steam-to-water halogen concentration ratio is in the range of 1×10^{-5} to 3×10^{-5}.

e. Water carryover in the main steam lines is assumed to be 0.1 percent of the total mass of steam transferred to the condenser. Measurements of the steam separation effected by the same types of separators used in this reactor vessel show that water carryover is less than 0.1 percent even at rated steam flow. The carryover fraction permits computation of the halogen activity carried to the main condenser in the water entrained in the steam.

f. None of the fission products released from the fuel is assumed to plate out.

The main steam line radiation monitors initiate closure of all main steam isolation valves when a preestablished radiation level is exceeded. This action prevents further transport of the fission products to the condenser. Using the listed assumptions, the following amounts of fission product activities are carried to the condenser:

Noble gases	8.0×10^{3} Ci
Iodine 131	4.2×10^{-1} Ci
Iodine 132	6.4×10^{-2} Ci
Iodine 133	2.2×10^{-1} Ci
Iodine 134	5.5×10^{-2} Ci
Iodine 135	1.4×10^{-1} Ci

14.11.2.6 Fission Product Release to Environs

The following assumptions and initial conditions were used in the calculation of fission product activity released to the environs (initial core):

a. The accident is assumed to occur while condenser vacuum is being maintained with the mechanical vacuum pump. During normal operation, vacuum is maintained with the steam-jet-air ejector, the discharge from which is through a holdup (time delay) and filter system. The assumed operation of the mechanical vacuum pump results in the discharge of the condenser activity directly to the environment via the elevated release point but without the benefits of holdup (decay) or filtration.

b. All of the noble gas activity transferred to the condenser during the assumed 10.5 second isolation valve closure time is assumed to be airborne in the condenser. The halogen activity transferred to the condenser experiences the removal effects of the condensate and forms an equilibrium condition between the condensate and the vapor volume.

c. The rate at which the condenser activity is discharged to the environment is dependent upon the free volume of the turbine and condenser, the volume of liquid in the condenser, and the discharge rate of the mechanical vacuum pump. The numerical values appropriate to these parameters are 208,000 ft^3 turbine plus condenser free volume, 12,500 ft^3 condenser liquid, and 1,800 cfm mechanical vacuum pump discharge rate.

d. If the mechanical vacuum pump is isolated, the activity released will be contained within the condenser. Due to the condenser air being at lower pressure than its surroundings, the only leakage, if any, would be inward. Therefore, no activity would be transported to the environs.

Based upon these conditions, the fission product release rate to the environment is shown in Table 14.11-l.

14.11.2.7 Radiological Effects

The radiological exposure resulting from the activity discharged to the environment (initial core) was determined for six meterological conditions. These conditions range from very stable to unstable and consider wind speeds of 1 meter/sec and 5 meters/sec. Table 14.11-2 shows that the maximum offsite exposure occurs at the site boundary, which is approximately 0.9 miles from the release point. The maximum radiological exposures at the site boundary are 1×10^{-3} rem thyroid and 1.2×10^{-2} rem whole body, which are well below the respective thyroid and whole body gamma reference values of 300 rem and 25 rem, respectively, set forth in 10 CFR 100. Due to the large flow rate of the mechanical vacuum pump, essentially all of the activity is exhausted to the environment in the 24-hour release period investigated. Therefore, a 30-day dose would be essentially equivalent to the dose obtained for the 24-hour period. NEDE-24011-P-A-9-US describes how the radiological effects of a CRDA have been affected by the change from 7 x 7 fuel to 8 x 8 fuel. The radiological effects are still orders of magnitude below those set forth in l0 CFR l00.

14.11.2.8 Elimination of the Main Steam Line High Radiation Signal

Upon detection of high radiation in the main steam lines, the main steam line radiation monitors (MSLRMs) originally performed the following actions: 1) scram the reactor, 2) close the main steam isolation valves, 3) close the Main Steam Line (MSL) drain isolation valves, 4) isolate the reactor coolant sample lines and 5) isolate and trip the condenser mechanical vacuum pump (MVP).

General Electric (GE) Licensing Topical Report, NEDO-31400A, October 1992, presents a generic bounding safety analysis which supports the removal of the automatic MSIV closure, MSL drain line isolation valve closure and the automatic reactor shutdown functions of the Main Steam Line Radiation Monitor (MSLRM). BFN has performed additional analyses as described below to justify eliminating the remaining trip/isolation functions of the MSLRMs. Eliminating the MSLRM automatic functions will reduce the potential for unnecessary reactor shutdowns and increase the plant operational flexibility. Following the elimination of these functions, the calculated radiological release consequences of the CRDA will not exceed the acceptable dose limits as specified in 10 CFR 100 and Standard Review Plan (SRP) 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)."

14.11.2.8.1 Summary of Revised Rod Drop Accident Analysis

The NEDO-31400A analyzes the consequences of the CRDA by assuming the CRDA source term carried away by the reactor steam is instantaneously deposited in the turbine and condenser and either leaks out of the condenser into the turbine building and then enters the environment or is released to the environment through the steam jet air ejector (SJAE) off-gas system via the plant stack. The bounding analysis is performed without the MSLRM automatic reactor shutdown and MSIV closure functions. The CRDA source term and radiological transport from the condenser are consistent with the standard approach outlined in the SRP 15.4.9. The NEDO-31400A generic results are presented in graphs such that site specific atmospheric dispersion coefficient (X/Q) values and offgas holdup times can be applied to determine the resulting CRDA doses.

Two additional analyses have been performed to demonstrate that the MVP trip/isolation function and the reactor coolant sample line isolation are not required in order to limit the consequences of an CRDA within acceptable values. Without the MVP trip/isolation, the CRDA source term in the condenser was assumed to be exhausted from the condenser at the MVP flow rate and released to the atmosphere via the plant stack. This release path goes directly to the stack with no holdup or filtering. Without the reactor coolant sample line isolation, the analysis assumes reactor coolant containing CRDA iodine source term is released into the secondary containment and is released to the environment via the standby gas treatment system (SGTS) to the plant stack. Fission products exiting the secondary containment prior to SGTS initiation are considered.

The results of these analyses are considered acceptable if the resulting doses are well within the 10 CFR 100 limits for offsite doses. SRP 15.4.9 defines "well within" as being below 25 percent of the 10 CFR 100 limits.

14.11.2.8.2 Application of NEDO-31400A to BFN

NEDO-31400A explicitly discusses the elimination of the automatic reactor shutdown and MSIV closure functions of the MSLRM. Since the MSL drain discharges to the condenser (as do the main steam lines), the NEDO analysis is also applicable to and bounds the elimination of the MSL drain isolation function. In order to apply the generic NEDO-31400A analysis to BFN, it must be demonstrated that the assumptions made and analysis performed bound those of BFN for a CRDA. The following is a comparison of the key input parameters used in the BFN CRDA analysis to demonstrate NEDO-31400A applicability.

Parameters	BFN	NEDO-31400A
Power	0.109 MW/Rod* (105%)	0.12 MW/Rod (105%)
Failed Fuel Rods	850 (NEDO 31400A and GESTAR II, NEDE-24011-P-A)	850
Operation	1000 days (FSAR Section 14.11.2.4(e))	Long Term
Releases from Fuel (non melted) (melted)	10% Noble 10% Iodine 100% Noble 50% Iodine	10% Noble 10% Iodine 100% Noble 50% Iodine
X/Q Ground (EAB)	1.22×10^{-4} sec/m^3 (FSAR Table 14.11-8)	2.5×10^{-3} sec/m^3
X/Q Fumigation (EAB)	2.4×10^{-5} sec/m^3	N/A
X/Q Elevated (EAB)	9.70×10^{-7} sec/m^3 (FSAR Table 14.11-8)	3.0×10^{-4} sec/m^3
Holdup (Delay Time)	7.3 days for Xenon, 9.7 hours for Kr (FSAR Section 9.5.4)	Graphs provided for various holdup times

*Calculated as: (3293 x 1.05 x 1.5) MW/ (764 x (64-2)) rods
= 0.109 MW/rod

Utilizing the BFN site specific parameters and the graphs provided in the NEDO-31400A analysis, the resulting BFN Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses for the condenser 1% per day leakage and off gas system release paths following a CRDA are well below the 10 CFR 100 limits and SRP 15.4.9 guidelines.

14.11.2.8.3 Elimination of Additional Main Steam Line Radiation Monitor Functions

The BFN MSLRM also isolates and trips the condenser mechanical vacuum pump (MVP) and isolates the reactor coolant sample lines. The NEDO-31400A assumes the MVP trips and is isolated such that the CRDA source term either leaks out of the condenser at 1% per day or is processed through the off-gas system with holdup volumes and charcoal filters. However, the MVP trip and isolation on high MSL radiation is not a safety-related function. This release path was not analyzed with the CRDA condenser source term in the NEDO or in the original BFN CRDA analysis. Therefore, the offsite dose impact with the CRDA source term being exhausted from the condenser via the MVP has been analyzed. The CRDA source term in the condenser as described in the NEDO was exhausted to the environment via the plant stack at the MVP flow rate of 1850 cfm. The flow in the stack was split between the base and top of the stack and the atmospheric dispersion coefficients were used as discussed in Section 14.11.3.6.f, g, and h. The resulting EAB and LPZ doses from the MVP release path are well below the 10 CFR 100 limits and the SRP 15.4.9 guidelines.

The release due to the elimination of the reactor water sample line isolation is another potential path that was not analyzed by the generic NEDO-31400A. This 3/4" sample line is connected to the discharge of a reactor recirculation pump but is normally isolated by its primary containment isolation valves. The line is used as an alternate means of obtaining samples for continuous conductivity monitoring of the reactor coolant. Thus, this line is normally closed unless the normal sample path from the RWCU demineralizers is out of service. The recirculation sample station is protected from overpressurization by pressure control valves, sample coolers, and relief valves (relief valves are in Unit 3 only). However, these overpressurization protection devices are not safety related. The samples are discharged directly to Radwaste.

If this alternate sample path is in operation at the time of a CRDA and the nonsafety-related overpressurization protection devices failed, the result could potentially overpressurize the instruments and produce a continuous blowdown of reactor coolant into the reactor building. This scenario is essentially a small break LOCA outside containment which is releasing CRDA source term directly to the secondary containment. Since the break is from a subcooled section of reactor coolant piping which is well below the reactor vessel normal water line, the analysis assumes that only CRDA source term iodines (i.e., no noble gases) exit through the break. This analysis demonstrates that this unlikely blowdown release to the reactor building would initiate isolation of secondary containment and start the standby gas treatment system. Fission products exiting the secondary containment prior to SGTS initiation have been considered and are treated as a ground level release. The fission products removed via SGTS were exhausted to the environment via the plant stack as per the assumptions in Section 14.11.3.6.b, c, e, f, g, and h. The resulting EAB and LPZ doses from the reactor coolant sample line release path are well below the 10 CFR 100 limits and the SRP 15.4.9 guidelines.

14.11.2.8.4 Radiological Effects of Eliminating the MSLRM Functions

The BFN analysis for the CRDA without MSLRM automatic reactor shutdown and isolation functions now consists of four potential release paths; condenser leakage at 1% per day into the turbine building or through SJAE and off-gas system as analyzed by the NEDO-31400A, and the MVP discharge and recirculation sample line discharge as analyzed in accordance with SRP 15.4.9. The "worst-case" radiological exposure resulting from the activity discharged from a CRDA and a SRP 15.4.9 source term would be from the MVP and recirculation sample line release paths combined. The combination of these paths maximizes the CRDA source term released and could occur simultaneously. The resulting combined EAB and LPZ doses from the MVP and reactor coolant sample line are well below the SRP 15.4.9 reference values of 75 REM thyroid and 6 REM whole body.

Based on the analyses above, the MVP, recirculation sample line, and MSL drain release paths have been analyzed and their isolation on a MSL high radiation signal is not required to mitigate the consequences of a CRDA. Units 2 and 3 have physically disconnected the MSLRM functions for automatic reactor shutdown, MSIV closure, MSL drain isolation, and recirculation sample line isolation but has retained the MSLRM function for MVP trip and isolation as an additional nonsafety- related preventative means of reducing the consequences of a CRDA. Unit 1 has not physically disconnected these functions.

14.11.3 Loss of Coolant Accident (LOCA)

Accidents that could result in release of radioactive material directly into the primary containment are the results of postulated nuclear system pipe breaks inside the drywell. All possibilities for pipe break sizes and locations have been investigated including the severance of small pipe lines, the main steam lines upstream and downstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop pipelines. This accident is established as the design basis loss of coolant accident.

Subsection 14.11.3 presents information on the analytical models used to analyze the LOCA for the initial operating cycle including the results of the analyses. This description is applicable only to the initial operating cycle but is generally applicable to analytical LOCA work performed for subsequent cycles. Additional information on

LOCA models currently in use is given in NEDO-20566[8] and NEDC-32484P.[9] Detailed plant specific information on models used and results of the LOCA analysis for the current operating cycle is given in a separate document prepared in conjunction with the reload licensing amendments. Additional information on the sequence of events during a LOCA and the response of the primary containment during a LOCA is given in NEDO-10320[10] and NEDC-32484P.[11]

14.11.3.1 Initial Conditions and Assumptions

The analysis of this accident is performed using the following assumptions:

a. The reactor is operating at the most severe condition at the time the recirculation pipe breaks, which maximizes the parameter of interest: primary containment response, fission product release or Core Standby Cooling System requirements.

b. A complete loss of normal AC power occurs simultaneously with the pipe break. This additional condition results in the longest delay time for the Core Standby Cooling Systems to become operational.

c. The recirculation loop pipeline is considered to be instantly severed. This results in the most rapid coolant loss and depressurization with coolant discharged from both ends of the break.

14.11.3.2 Nuclear System Depressurization and Core Heatup

In Section 6, "Core Standby Cooling Systems," the initial phases of the loss of coolant accident are described and evaluated. Included in that description are the rapid depressurization of the nuclear system, the operating sequences of the Core Standby Cooling Systems, the heatup of the fuel, and the perforation of fuel rods. Analysis shows that a maximum of 9.0 percent of the fuel rods reach the pressure and temperature conditions necessary for perforation.

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[8] General Electric Company Analytical Model for Loss-of Coolant Analysis in Accordance with 10CFR50 Appendix K, NEDO-20566.

[9] General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P.

[10] The General Electric Pressure Suppression Containment Analytical Model, NEDO-10320.

[11] General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P.

14.11.3.3 Primary Containment Response

14.11.3.3.1 Initial Conditions and Assumptions

The following assumptions and initial conditions were used in calculating the effects of a loss of coolant accident on the primary containment. (These assumptions are in addition to those specified for the loss of coolant accident described in paragraph 14.11.3.1.)

a. The reactor is assumed to be operating at the maximum possible steady-state power level and pressure at the time the accident occurs. This maximizes the reactor pressure during the blowdown which in turn maximizes the blowdown flow rate.

b. The break area through which the reactor coolant can escape to the drywell is maximized by assuming the reactor is operating on one recirculation loop with the equalizer valves open. In this configuration, mass escapes from the reactor vessel via the broken loop as well as from jet pump backflow from the unbroken loop through the equalizer valves to the broken loop. This results in the most severe primary containment pressure transient. For the equalizer line to be open, an interlock requires the reactor to be operating on only one recirculation pump with the idle pump's discharge valve closed. The maximum power level under this condition is approximately 80 percent. It is recognized that this assumption is inconsistent with the assumption regarding initial reactor power but is used to maximize the break area. It is also recognized that this assumption is conservative for Unit 3 since the recirculation ring header has been split into two independent halves and the equalizer valves removed. Removal of the equalizer valves prevents the cross flow from the unbroken loop and thus reduces the break effluent.

c. The reactor is assumed to go subcritical at the time of accident initiation due to void formation in the core region. Scram also occurs in less than one second from receipt of the high drywell pressure and low water level signals, but the difference in shutdown time between zero and one second is negligible.

d. The sensible heat released in cooling the fuel to 545°F (the normal primary system operating temperature) and the core decay heat were included in the reactor vessel depressurization calculation. The rate of energy release was calculated using a conservatively high heat transfer coefficient throughout the depressurization. Because of this assumed high energy release rate the vessel is maintained at near rated pressure about ten seconds. The high vessel pressure increases the calculated flow rates out of the break; this is conservative for containment analysis purposes. With the vessel fluid temperature remaining near 545°F, however, the release of sensible energy stored below 545°F is negligible during the first ten seconds. The later release of this sensible energy does not affect the peak drywell pressure. The small effect of this energy on the end-of-transient pressure suppression pool temperature is included in the calculations.

e. The main steam isolation valves were assumed to start closing at 0.5 seconds after the accident, and the valves were assumed to be fully closed in the shortest possible time of three seconds following closure initiation. Actually, the closures of the main steam isolation valves are expected to be the result of low water level, so these valves may not receive a signal to close for over four seconds, and the closing time could be as high as 10 seconds. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure which maximizes the discharge of high energy steam and water into the primary containment.

f. The feedwater flow was assumed to stop instantaneously at time zero. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

g. The vessel depressurization flow rates were calculated using Moody's critical flow model[12] assuming "liquid only" outflow because this maximizes the energy release to the containment. "Liquid only" outflow means that all vapor formed in the vessel due to bulk flashing rises to the surface rather than being entrained in the exiting flow.

Some entrainment of the vapor would occur and would significantly reduce the reactor vessel discharge flow rates. Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. However, actual flow rates through larger flow areas are less than the model indicates due to the effects of a near homogeneous two-phase flow pattern and phase nonequilibrium. These effects are in addition to the reduction due to vapor entrainment discussed above.

h. The pressure response of the containment is calculated assuming:

1. Thermodynamic equilibrium in the drywell and pressure suppression chamber. Because complete mixing is nearly achieved, the error introduced by assuming complete mixing is negligible and in the conservative direction.

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[12] Moody, F. J. "Maximum Flow of a Rate Single Component Two-Phase Mixture," Journal of Heat Transfer ASME Series C, Vol 83, p. 134.

2. The constituents of the fluid flowing in the drywell to pressure suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the drywell vents. Actually, some of the liquid will remain behind in a pool on the drywell floor so that the calculated drywell pressure is conservatively high.

3. The flow in the drywell pressure suppression pool vents is compressible except for the liquid phase.

4. No heat loss from the gases inside the primary containment is assumed.

i. The initial conditions within the containment assumed for the analysis were:

Drywell	
Pressure, psig	0.75
Temperature, °F	135
Humidity, percent	20
Pressure Suppression Chamber	
Pressure, psig	0.75
Water Temperature, °F	95
Humidity, percent	100

14.11.3.3.2 Containment Response

The calculated pressure and temperature responses of the containment are shown in Figures 14.11-10 and 14.11-11. Figure 14.11-10 shows that the calculated drywell peak pressure is 49.6 psig, which is well below the maximum allowable pressure of 62 psig. After the discharge of the primary coolant from the reactor vessel into the drywell, the temperature of the pressure suppression chamber water approaches 170°F (Figure 14.11-12), and the primary containment pressure stabilizes at about 27 psig, as shown on Figure 14.11-10. Most of the noncondensible gases are forced into the pressure suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the pressure suppression chamber via the vacuum breaker system as the drywell pressure decreases due to steam condensation. The Core Spray System removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to less than 0.1 percent. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the pressure suppression chamber via the drywell-to-pressure suppression chamber vent pipes. Steam flow is negligible. The energy transported to the pressure suppression chamber water is then removed from the primary containment system by the RHRS heat exchangers.

Prior to activation of the RHRS containment cooling mode (arbitrarily assumed at 600 seconds after the accident), the RHRS pumps (LPCI mode) have been adding liquid to the reactor vessel. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow offers considerable cooling to the drywell and causes a depressurization of the containment as the steam in the drywell is condensed. At 600 seconds, the RHRS pumps are assumed to be switched from the LPCI mode to the containment cooling mode. The containment spray would normally not be activated at all and the changeover to the containment cooling mode need not be made for several hours. There is considerable time available to place the containment cooling system in operation because about eight hours will pass before the maximum allowable pressure is reached with no containment cooling.

To assess the primary containment long term response after the accident, an analysis was made of the effects of various containment spray and containment cooling combinations. For all cases, one of the core spray loops is assumed to be in operation. The long term pressure and temperature response of the primary containment was analyzed for the following RHRS containment cooling mode conditions:

Case A Operation of both RHRS cooling loops - four RHRS pumps, four service water pumps, and four RHRS heat exchangers - with containment spray.

Case B Operation of two RHRS cooling loops with one RHRS pump, one service water pump, and one RHRS heat exchanger on each loop - with containment spray.

Case C Operation of one RHRS cooling loop with two RHRS pumps, two service water pumps, and two RHRS heat exchangers - with containment spray.

The initial pressure response of the containment (the first 30 seconds after break) is the same for each of the above conditions. During the long term containment response (after depressurization of the reactor vessel is complete), the pressure suppression pool is assumed to be the only heat sink in the containment system. The effects of decay energy, stored energy, and energy from the metal-water reaction on the pressure suppression pool temperature are considered.

Case A

This case assumes that both RHRS loops are operating in the containment cooling mode. This includes four RHRS heat exchangers, four RHRS pumps, and four RHR service water pumps. The RHRS pumps draw suction from the pressure suppression pool and pump water through the RHRS heat exchangers and into the drywell as containment spray. This forms a closed cooling loop with the pressure suppression pool. This pressure suppression pool cooling condition is arbitrarily assumed to start at 600 seconds after the accident. Prior to this time the RHRS pumps are used to flood the core (LPCI mode).

The containment pressure response to this set of conditions is shown as curve "a" in Figure 14.11-10. The corresponding drywell and pressure suppression pool temperature responses are shown as curves "a" in Figures 14.11-11 and 14.11-12. After the initial rapid temperature rise in the containment. When the energy removal rate of the RHRS exceeds the energy addition rate from the decay heat, the containment pressure and temperature decrease to their preaccident values. Table 14.11-3 summarizes the cooling equipment operation, the peak containment pressure following the initial blowdown peak, and the peak pressure suppression pool temperature.

Case B

This case assumes that both RHRS loops are operating in the containment cooling mode. However, only one RHR heat exchanger, one RHR pump, and one RHR service water pump on each loop are assumed to be in operation. As in the previous case, the RHRS containment cooling mode is assumed to be activated at 600 seconds after the accident. The containment pressure response to this set of conditions is shown as curve "b" in Figure 14.11-10. The corresponding drywell and pressure suppression pool temperature responses are shown as curves "b" in Figures 14.11-11 and 14.11-12. A summary of this case is shown in Table 14.11-3.

Case C

This case assumes that one RHRS loop is operating in the containment cooling mode. This includes two RHRS heat exchangers, two RHRS pumps, and two RHR service water pumps.

This case represents the most degraded condition of heat removal while in the containment cooling mode. It is assumed that this condition is established at 600 seconds after the accident.

The containment response to this set of conditions is shown as curve "c" in Figure 14.11-10. The corresponding drywell and pressure suppression pool temperatures are shown as curves "c" in Figures 14.11-11 and 14.11-12. A summary of this case is shown in Table 14.11-3. Case C pressure suppression pool responses have been reanalyzed by NEDC-32484P, Revision 2, and GE-NE-B13-01755-2, Revision 2.

14.11.3.3.3 Metal Water Reaction Effects on the Primary Containment

If Zircaloy in the reactor core is heated to temperatures above about 2000°F in the presence of steam, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied with an energy release of about 2800 Btu per pound of zirconium reacted. The energy produced is accommodated in the pressure suppression chamber pool. The hydrogen formed, however, will result in an increased drywell pressure due simply to the added volume of gas to the fixed containment volume. Although very small quantities of hydrogen are produced during the accident, the containment has the inherent ability to accommodate a much larger amount as discussed below.

The basic approach to evaluating the capability of a containment system with a given containment spray design is to assume that the energy and gas are liberated from the reactor vessel over some time period. The rate of energy release over the entire duration of the release is arbitrarily taken as uniform, since the capability curve serves as a capability index only, and is not based on any given set of accident conditions as an accident performance evaluation might be.

It is conservatively assumed that the pressure suppression pool is the only body in the system which is capable of storing energy. The considerable amount of energy storage which would take place in the various structures of the containment is neglected. Hence, as energy is released from the core region, it is absorbed by the pressure suppression pool. Energy is removed from the pool by heat exchangers which reject heat to the service water. Because the energy release is taken as uniform and the service-water temperature and exchanger flow rate are constant, the temperature response of the pool can be determined. It is assumed that the pressure suppression chamber gases are at the pressure suppression chamber water temperature.

The metal-water reaction during core heatup is calculated by the core heat-up mode described in Subsection 14.8. The extent of the metal-water reaction thus calculated is less than 0.1 percent of all the zirconium in the core. As an index of the containment's ability to tolerate postulated metal-water reactions, the concept of "Containment Capability" is used. Since this capability depends on the time domain, the duration over which the metal-water reaction is postulated to occur is one of the parameters used.

Containment capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction during a specified duration without exceeding the maximum allowable pressure of the containment. To evaluate the containment capability, various percentages of metal-water reaction are assumed to take place over certain time period. This analysis presents a method of measuring system capability without requiring prediction of the detailed events in a particular accident condition.

Since the percent metal-water reaction capability varies with the duration of the uniform energy and gas release, the percent metal-water reaction capability is plotted against the duration of release. This constitutes the containment capability curves as shown in Figure 14.11-14. All points below the curves represent a given metal-water reaction and a given duration which will result in a containment peak pressure which is below the maximum allowable pressure. The calculations are made at the end of the energy release duration because the number of moles of gases in the system in then at a maximum, and the pressure suppression pool temperature is higher at this time than at any other time during the energy release.

It should be noted that the curves are actually derived from separate calculations of two conditions: the "steaming" and the "nonsteaming" situation. The minimum amount of metal-water reaction which the containment can tolerate for a given duration is given by the condition where all of the noncondensible gases are stored in the pressure suppression chamber. This condition assumes that "steaming" from the drywell to the pressure suppression chamber results in washing all of the noncondensible gases into the pressure suppression chamber. This is shown as the flat portion of the containment capability characteristic curve. Activation of containment sprays condense the drywell steam so that no steaming occurs, thus allowing noncondensibles to also be stored in the drywell. This is denoted by the rising (spray) curve. The intersection between the no spray curve and the spray curve represents the duration and metal water reaction energy release which just raises all the spray water to the saturation temperature at the maximum allowable containment pressures.

For durations to the left of the intersection, some steam is generated and all the gases are stored in the pressure suppression chamber. For durations to the right of the intersection, the spray flow is subcooled as it exits from drywell by increasing amounts as the duration is increased.

The energy release rate to the containment is calculated as follows:

$$q_{IN} = \frac{Q_O + Q_{MW} + Q_S}{T_D}$$

where:

q_{IN} = Arbitrary energy release rate to the containment Btu per sec,

Q_O = Integral of decay power over selected duration of energy gas release, Btu,

Q_{MW} = Total chemical energy released exothermically from selected metal-water reaction, Btu,

Q_S = Initial internal sensible energy of core fuel and cladding, Btu. and

T_D = Selected duration of energy and gas release, seconds.

The total chemical energy released from the metal-water reaction is proportional to the percent metal-water reaction. The initial internal sensible energy of the core is taken as the difference between the energy in the core after the blowdown and the energy in the core at a datum temperature of 250°F.

The temperature of the drywell gas is found by considering an energy balance on the spray flows through the drywell as described in Subsection 14.8.

Based upon the drywell gas temperature, pressure suppression chamber gas temperature and the total number of moles in the system, as calculated above, the containment pressure is determined. The containment capability curves in Figure 14.11-14 present the results of the parametric investigation.

14.11.3.4 Fission Products Released to Primary Containment

The following assumptions and initial conditions were used in calculating the amounts of fission products released from the nuclear system to the drywell:

a. Source terms based on TID 14844 methodology. These source terms are generally comparable to those based on the methodology utilized by the ORIGEN Code.

b. The reactor has been operating at design power (3458 MWt) for 1,000 days prior to the accident. This is appropriate for irradiation times up to 1400 days as noted by calculations performed utilizing the ORIGEN Code.

c. One hundred percent of the equilibrium radioactive noble gas inventory developed as a result of such operation is released.

d. Twenty-five percent of the equilibrium radioactive iodine inventory developed as a result of such operation is released. Of this 25 percent, 91 percent is assumed to be elemental iodine, 5 percent in particulate form, and 4 percent in the form of organic iodides. Table 14.11-4 gives the inventory of each isotope in the primary containment available for leakage.

14.11.3.5 Fission Product Release From Primary Containment

Fission products are released from the primary containment to the secondary containment via primary containment penetration leakage at the Technical Specification leakage limit. The following assumptions were used in calculating the amounts of fission products released from the primary containment:

a. The primary containment free volume is 283,000 ft^3.

b. The primary to secondary containment leak rate was taken as two percent volume per day (235 cfh).

14.11.3.6 Fission Product Release to Environs

Secondary Containment Releases

The fission product activity in the secondary containment at any time (t) is a function of the leakage rate from the primary containment, the volumetric discharge rate from the secondary containment and radioactive decay. During normal power operation, the secondary containment ventilation rate is 75 air changes per day; however, the normal ventilation system is turned off and the Standby Gas Treatment System (SGTS) is initiated as a result of low reactor water level, high drywell pressure, or high radiation in the Reactor Building. Any fission product removal effects in the secondary containment such as plateout are neglected. The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment and the efficiency of the various components of the SGTS.

The following assumptions were used to calculate the fission product activity released to the environment from the secondary containment:

a. The leakage from primary containment to secondary containment mixes instantaneously and uniformly within the secondary containment.

b. The effective volume of the secondary containment is 50 percent of the total free volume of a single reactor zone plus 50 percent of the refueling zone. The resulting effective secondary containment volume is 1,931,502 ft^3.

c. The SGTS removes fission products from secondary containment. If only two of the SGTS trains are in operation (i.e., SGTS flow of 16,200 cfm), a short period exists at the start of the accident during which the secondary containment becomes pressurized relative to the outside environment.

During this short time period, a very small amount of secondary containment atmosphere (~35 ft^3) will be released directly to the environment unfiltered from the Reactor Building. Once the secondary containment pressure is reduced below atmospheric pressure, all releases from secondary containment to the environment are through the SGTS filters via the plant stack. If all three trains of SGTS are in operation (i.e., SGTS flow of 22,000 cfm), all releases to the environment from secondary containment are through the SGTS filters via the plant stack.

d. The Containment Atmospheric Dilution (CAD) system operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post accident. This flow is filtered via the SGTS filters.

e. Filter efficiency for the SGTS was taken as 90 percent for organic and inorganic (elemental) iodine.

f. Release to the environment from the plant stack is composed of two flow paths. A continuous ground level release of 10 cfm occurs at the base of the stack. This flow results from leakage through the backdraft dampers in the base of the stack. Subsection 5.3.3, "Secondary Containment System Description" describes the backdraft dampers. The 10 cfm leakage mixes uniformly within the rooms at the base of the stack (34,560 ft^3). The remaining SGTS flow exits the stack at a height of 183 meters above ground elevation.

g. Fumigation conditions exist for the first 30 minutes post accident.

h. Atmospheric dispersion coefficients, X/Q, for elevated releases under fumigation conditions, elevated releases under normal atmospheric conditions and ground level releases at the base of the stack are used. X/Q values applicable to the time periods, distances and geometric relationships (offsite and control room) are shown in Table 14.11-8.

14.11.3.7 Radiological Effects

The LOCA provides the most severe radiological releases to the primary and secondary containments and thus serves as the bounding design basis accident in determining post-accident offsite and control room personnel doses.

Offsite Doses

Offsite doses of interest resulting from the activity released to the environment as a consequence of the loss of coolant accident are the 2-hour whole body gamma dose, beta dose and the thyroid inhalation dose at the site boundary (1,465 meters), and the corresponding 30-day doses at the low population zone (LPZ) boundary 3,200 meters).

The offsite doses are calculated using a combination of the STP and FENCEDOSE computer programs. The STP program models the fission product transport from the primary containment to release to the environment. The model accounts for fission product decay, flow rates, filter absorption, dilution, release rates and release points. The FENCEDOSE computer program models the atmospheric dispersion to the offsite receptor points by use of appropriate X/Qs and calculates the gamma, beta, and thyroid doses.

The largest calculated total offsite dose is well within the 10 CFR 100 guideline values.

Control Room

The control room doses are calculated using a combination of the STP and COROD computer programs. The STP program models the fission product transport from the primary containment to release to the environment. The model accounts for fission product decay, flow rates, filter absorption, dilution, release rates, and release points. The COROD computer program accounts for the atmospheric dispersion to the control room intakes by use of appropriate X/Qs and models the control bay habitability zone filtered pressurization flow, unfiltered inleakage, occupancy times, breathing rates and calculates the gamma, beta, and thyroid doses. Atmospheric dispersion coefficients are based on release point, geometric relationship of the release point and receptor and atmospheric conditions based on site specific meteorological data. The COROD computer code calculates the gamma dose by a typical point-kernel methodology accounting for the control room geometry. The thyroid dose was reduced by ratioing to the ICRP-30 conversion factors. This resulted in a reduction factor of 1.7 for the dose for the 0 to 30 minute time frame and a factor of 1.35 for times after 30 minutes.

The direct gamma dose contribution from the piping inside secondary containment, the secondary containment atmosphere and the cloud dose are included. One section of core spray piping in each unit is routed just outside the common Control Building/Reactor Building wall. This piping will be carrying pressure suppression chamber water in the event of a LOCA.

All of these exposure mechanisms (filtered pressurization flow, unfiltered inleakage, cloud dose and direct dose) are combined to produce a total control room dose for the duration of the accident. It was determined that the differences between the case with two SGTS fans in operation with a small amount of unfiltered secondary containment release and the case with three SGTS fans in operation with all releases being filtered and via the plant stack are negligible. The 30 day integrated post-accident doses in the control room are within the limits of 5 REM whole body gamma dose, 30 REM beta and 30 REM to the thyroid as specified in 10 CFR 50, Appendix A General Design Criteria 19.

The Committed Effective Dose Equivalent (CEDE) for the thyroid plus the whole body gamma Deep Dose Equivalent (DDE) is below the 5 REM Total Effective Dose Equivalent (TEDE) limit.

14.11.4 Refueling Accident

The current safety evaluation for the Refueling Accident is contained in the licensing topical report for nuclear fuel, "General Electric Standard Application For Reactor Fuel," NEDE-24011-P-A, and subsequent revisions thereto. Accidents that result in the release of radioactive materials directly to the secondary containment are events that can occur when the primary containment is open. A survey of the various plant conditions that could exist when the primary containment is open reveals that the greatest potential for the release of radioactive material exists when the primary containment head and reactor vessel head have been removed. With the primary containment open and the reactor vessel head off, radioactive material released as a result of fuel failure is available for transport directly to the reactor building.

Various mechanisms for fuel failure under this condition have been investigated. Refueling Interlocks will prevent any condition which could lead to inadvertent criticality due to control rod withdrawal error during refueling operations when the mode switch is in the Refuel position. The Reactor Protection System is capable of initiating a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during deliberate criticality tests with the reactor vessel head off. The possibility of mechanically damaging the fuel has been investigated.

The design basis accident for this case is one in which one fuel assembly is assumed to fall onto the top of the reactor core.

The discussion in Subsection 14.11.4.1 applies to the dropping of a 8 x 8 assembly. The analyses for all current General Electric product line fuel bundle designs are contained in supplements to NEDE-24011-P-A. The NEDE evaluates each new fuel design against the 7x7 fuel design for the original core load. The 7x7 fuel handling accident resulted in 111 failed fuel rods. For the 8x8 fuel design, the activity released due to a fuel handling accident will be less than 84% of the activity released by the original 7x7 fuel design. For the 9x9 fuel design the activity will be less than 83.5% of the activity released by the original 7x7 fuel design. The historical and current calculated doses are much less than the regulatory guidelines.

14.11.4.1 Assumptions

1. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

2. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the reactor core and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple, or the grapple cable breaks.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

14.11.4.2 Fuel Damage

Dropping a fuel assembly onto the reactor core from the maximum height allowed by the refueling equipment, less than 30 feet, results in an impact velocity of 40 ft/sec. The kinetic energy acquired by the falling fuel assembly is approximately 18,150 ft-lb and is dissipated in one or more impacts. The first impact is expected to dissipate most of the energy and cause the largest number of cladding failures. To estimate the expected number of failed fuel rods in each impact, an energy approach has been used.

The fuel assembly is expected to impact on the reactor core at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. Fuel rods are expected to absorb little energy prior to failure due to bending, if it is assumed that each fuel rod resists the imposed bending load by two equal, opposite concentrated forces. Actual bending tests with concentrated point loads show that each fuel rod absorbs about 1 ft-lb prior to cladding failure. For rods which fail due to gross compression distortion, each rod is expected to absorb about 250 ft-lbs before cladding failure (this is based on 1 percent uniform plastic deformation of the rods). The energy of the dropped assembly is absorbed by the fuel, cladding, and other core structure. A fuel assembly consists of about 72 percent fuel, 11 percent cladding, and 17 percent other structural material by weight. Thus, the assumption that no energy is absorbed by the fuel material inserts considerable conservatism into the mass-energy calculations that follow.

The energy absorption on successive impacts is estimated by consideration of a plastic impact. Conservation of momentum under a plastic impact show that the fractional kinetic energy absorbed during impact is where

$$1 - \frac{M_1}{M_1 + M_2}$$

M_1 is the impacting mass and M_2 is the struck mass. Based on the fuel geometry within the reactor core, four fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is about 80 percent.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 24 more fuel assemblies, so that after the second impact only 135 ft-lbs (about 1 percent of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rods fail on a third impact.

If the dropped fuel assembly strikes only one or two fuel assemblies on the first impact, the energy absorption by the core support structure results in about the same energy dissipation on the first impact as in the case where four fuel assemblies are struck. The energy relations on the second and third impacts remain about the same as in the original case. Thus, the calculated energy dissipation is as following:

First impact	80 percent
Second impact	19 percent
Third impact	1 percent (no cladding failures)

The first impact dissipates 0.80 x 18,150 or 14,500 ft-lbs of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure, and because 1 ft-lb of energy is sufficient to cause cladding failure due to bending, all 62 rods of the dropped fuel assembly are assumed to fail. Because the 8 tie rods of each struck fuel assembly are more susceptible to bending failure than the other 54 rods, it is assumed that they fail upon the first impact. Thus 4 x 8 = 32 tie rods (total in four assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod due to compression, 250 ft-lbs of energy is required. To cause failure of all the remaining rods of the four struck assemblies, 250 x 54 x 4 or 54,000 ft-lbs of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures due to compression is computed as follows:

$$\frac{0.5 \times 14{,}500 \times \left(\frac{11}{11 + 17}\right)}{250} = 12$$

Thus, during the first impact, the fuel rod failures are as follows:

Dropped assembly	-	62	rods (bending)
Struck assemblies	-	32	tie rods (bending)
Struck assemblies	-	12	rods (compression)
		106	failed rods

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 24 struck assemblies are the tie rods subjected to bending failure. Thus, 2 x 8 = 16 tie rods are assumed to fail. The number of fuel rod failures due to compression on the second impact is computed as follows:

$$\frac{\frac{0.19}{2} \times 18{,}150 \times \frac{11}{11 + 17}}{250} = 3$$

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	-	16	tie rods (bending)
Struck assemblies	-	3	rods (compression)
		19	failed rods

The total number of failed rods (GE 8x8 fuel design) resulting from the accident is as follows:

First impact	-	106	rods
Second impact	-	19	rods
Third impact	-	0	rods
		125	failed rods (total)

14.11.4.3 <u>Fission Product Release From Fuel</u>

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

a. The reactor fuel has an average irradiation time of 1000 days at design power up to 24 hours prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not significantly increase the concentration of the fission products of concern. The 24-hour decay time allows time for the reactor to be shut down, the nuclear system depressurized, the reactor vessel head removed, and the reactor vessel upper internals removed. It is not expected that these evolutions could be accomplished in less than 24 hours.

b. The activity in the fuel bundle is determined from

$$q(Ci) = \frac{0.865\ x\ 10^6\ x\ f}{n} \times P_o \gamma_i (1 - e^{-\lambda_i T_o}) \times (e^{\lambda_i t_D})$$

where

f	=	peaking factor, taken as 1.5
n	=	number of fuel bundles in core (n = 764)
P_o	=	thermal power level (P_o = 3458 MWt)
γ_i	=	fission yield for isotope i
λ_i	=	decay constant of isotope i
T_o	=	residence time in core ($T_o = 8.64 \times 10^7$ sec)
t_D	=	decay time between shutdown and removal of the vessel head (24 hrs)

c. Due to the negligible particulate activity available for release in the fuel plenums or from the unmelted fuel, none of the solid fission products is assumed to be released from the fuel.

d. One hundred twenty-five fuel rods are assumed to fail. This was the conclusion of the analysis of mechanical damage to the fuel based on the GE 8x8 fuel design.

14.11.4.4 Fission Product Release to Secondary Containment

The following assumptions were used to calculate the fission product release to the secondary containment:

a. Fraction of Fuel Rod Inventory Released

Noble Gases (Except Kr 85)	10 percent
Kr 85	30 percent
Iodines	10 percent

b. Iodine Decontamination Factor in Reactor Cavity Pool Water — 100

14.11.4.5 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the dose existing at the exclusion area boundary and at the low population zone due to fission product release.

a. High radiation levels in the reactor building will isolate the normal ventilation system and actuate the Standby Gas Treatment System. The isolation dampers were assumed to close in 15 seconds.

b. The relative humidity in the secondary containment is 70 percent. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident, except for the addition of the released fission products. The relative humidity in the secondary containment will therefore be considerably below any levels which may be detrimental to the filter media in the Standby Gas Treatment System. However, as mentioned previously, the charcoal beds and absolute filter media, as well as the air flowing through the filter system, are heated 5°F above the mixture entering the system, reducing the relative humidity to 70 percent or less.

c.	Standby Gas Treatment System Filter Efficiency	0.90
d.	Height of the Main Stack	183 meters
e.	Distance to Exclusion Area Boundary	1,300 meters
f.	Distance to Low Population Zone	3,200 meters
g.	Mixing Air Volume	4,900 FT^3
h.	Ventilation Air Flow Prior to Damper Isolation	22,000 CFM

The design basis fuel handling accident assumes that during the refueling period a fuel bundle is dropped into the reactor cavity pool. The dropped fuel bundle strikes additional bundles in the reactor core fracturing 125 fuel pins (assuming GE 8x8 fuel design). Ten percent of the halogen isotopes inventory plus 10 percent of all noble gases inventory (except Kr 85 which is 30 percent of this inventory) will be released from the fractured fuel rods. An overall effective decontamination factor of 100 is applicable for iodine released at depth under water. The radioactive releases to the air space above the pool are released through the refueling zone ventilation and the Standby Gas Treatment Systems. The assumptions used to evaluate the fuel handling design basis accident event are defined in Nuclear Regulatory Commissions Regulatory Guide 1.25. Further guidance is contained in the standard review plans in NUREG-800, Section 15.7.4.

In order to evaluate the effect of refueling zone ventilation damper closure time, the analysis includes doses from air bypassing the Standby Gas Treatment System.

The bypass is occurring through the Refueling Zone Ventilation System. For this evaluation, it is assumed that the portion of the ventilation system dedicated to the reactor vessel pool and the spent fuel storage pool provides the bypass flow. The gases released from the damaged fuel bundles are assumed to be confined to an air volume bounded by the perimeter of the pool and mixed to a height of no more than 4 feet above the pool. The activity released to the environment before the dampers close is taken from the air volume over the pool expelled through the ventilation system. The total activity released is greater for a fuel handling accident in the reactor cavity pool than for an accident in the fuel storage pool. Normally, the number of fuel rods fractured in a drop into the reactor vessel pool is slightly larger than the number of rods fractured in a drop into the storage pool. This provides a bigger source for the vessel event. However, the ventilation flow from the storage pool area is twice the size of the flow from the reactor vessel area. The difference in flows transports more activity to the environment in a given time period. Therefore, for conservatism the number of rods damaged and resulting activity released is based on a fuel handling accident in the reactor cavity, and the mixing volume and ventilation is based on a release over the spent fuel pool.

The bypass flow not only bypasses the SGTS filters, it is also released from a roof vent rather than the main stack. The atmospheric dispersion, X/Q, of releases from the top of the stack is significantly smaller than the atmospheric dispersion factors for the roof vent releases. The result of this change is to make the dose contribution from the roof vent releases more important than if all releases were through the stack. Almost all the dose is from the roof vent release.

The fuel handling accident was evaluated using the STP, FENCEDOSE, and COROD computer programs described in Section 14.11.3.7. The calculations simulate an initial time period without filtration of the releases. Following the initial time period, the releases are filtered. Computations were prepared with an atmospheric dispersion, X/Q, for elevated releases and with X/Q data for ground level releases appropriate for the EAB and LPZ boundaries. The final dose evaluations become the dose contributions from the initial ground level release plus the contribution from the release of the balance of the activity through the stack (base and top).

14.11.4.6 Radiological Effects

The radiological exposures following the refueling accident have been evaluated at the site boundary and at the LPZ boundary. The calculated dose assumes that the bypass activity is exhausted through a roof vent and, after the dampers close, the activity is processed through the SGTS and the plant stack.

Boundary dose resulting from design basis accident events has been judged by comparing the dose to the dose in 10 CFR 100, Reactor Site Criteria. This regulation uses radiation doses of 300 rem to the thyroid and 25 rem whole body as guides for doses to the public under accident conditions. Fuel handling accidents in the past have been judged as having acceptable consequences if the dose is a small part of 10 CFR 100. In the standard review plan, NUREG-800, a small part has been defined as 25 percent. The calculated doses are much less than the guidelines.

14.11.5 Main Steam Line Break Accident

Accidents that result in the release of radioactive materials outside the secondary containment are the results of postulated breaches in the nuclear system process barrier. The design basis accident is a complete severance of one main steam line outside the secondary containment. Figure 14.11-15 shows the break location. The analysis of the accident is described in three parts as follows:

a. Nuclear System Transient Effects

   This includes analysis of the changes in nuclear system parameters pertinent to fuel performance and the determination of fuel damage.

b. Radioactive Material Release

   This includes determination of the quantity and type of radioactive material released through the pipe break and to the environs.

c. Radiological Effects

   This portion determines the dose effects of the accident to offsite persons.

14.11.5.1 Nuclear System Transient Effects

14.11.5.1.1 Assumptions

The following assumptions are used in evaluating response of nuclear system parameters to the steam line break accident outside the secondary containment:

a. The reactor is operating at design power.

b. Reactor vessel water level is normal for initial power level assumed at the time the break occurs.

c. Nuclear system pressure is normal for the initial power level.

d. The steam pipeline is assumed to be instantly severed by a circumferential break. The break is physically arranged so that the coolant discharge through the break is unobstructed. These assumptions result in the most severe depressurization rate of the nuclear system.

e. For the purpose of fuel performance, the main steam isolation valves are assumed to be closed 10.5 seconds after the break. This assumption is based on the 0.5 second time required for the development of the automatic isolation signal (high differential pressure across the main steam line flow restrictor) and the 10-second closure time for the valves.

For the purpose of radiological dose calculations, the main steam isolation valves are assumed to be closed at 5.5 seconds after the break. Faster main steam isolation valve closure could reduce the mass loss until finally some other process line break would become controlling. However, the resulting radiological dose for this break would be less than the main steam line break with a five-second valve closure. Thus, the postulated main steam line break outside the primary containment with a five-second isolation valve closure results in maximum calculated radiological dose and is therefore the design basis accident.

f. The mass flow rate through the upstream side of the break is assumed to be not affected by isolation valve closure until the isolation valves are closed far enough to establish limiting critical flow at the valve location. After limiting critical flow is established at the isolation valve, the mass flow is assumed to decrease linearly as the valve is closed.

g. The mass flow rate through the downstream side of the break is assumed to be not affected by the closure of the isolation valves in the unbroken steam lines until those valves are far enough closed to establish limiting critical flow at the valves. After limiting critical flow is established at the isolation valve positions, the mass flow is assumed to decrease linearly as the valves close.

h. In calculating the rate of water level rise inside the vessel, it is assumed that the steam bubbles formed during depressurization rise at an average velocity of about 1 foot per second relative to the liquid. This assumption is predicted by analysis[13] and confirmed experimentally.[14]

i. After the level of the mixture inside the reactor vessel rises to the top of the steam dryers, the quality of the two-phase mixture discharged through the break is assumed constant at its minimum value. This assumption maximizes

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[13] Moody, F. J.: "Liquid/Vapor Action In a Vessel During Blowdown" APED-5177, June 1966, Wilson, J.F., et al: "The Velocity of Rising Steam In A Bubbling Two-Phase Mixture," ANS Transation, Vol 5, No. 1, Page 151 (1962).

[14] Ianna, P.W., et al: "Design and Operating Experience Of The ESADA Vallecitos Experimental Superheat Reactor (Eversr)";

the total mass of coolant discharged through the break because most of the mixture flow will actually be at a higher quality.

j. Feedwater flow is assumed to decrease linearly to zero over the first four seconds to account for the slowing down of the turbine-driven feedpumps in response to the rise in reactor vessel water level.

k. A loss of auxiliary AC power is assumed to occur simultaneous with the break. This results in the immediate loss of power to the recirculation pumps. Recirculation flow is assumed to coast down according to momentum computations for the recirculation system.

l. Recirculation system drive pump head is assumed to be zero when the coolant at the pump suction reaches 1 percent quality. This assumption accounts for the effects of cavitation on recirculation drive pump capacity as the pumps coast down.

14.11.5.1.2 Sequence of Events

The sequence of events following the postulated main steam line break is as follows:

The steam flow through both ends of the break increases to the value limited by critical flow considerations. The flow from the upstream side of the break is limited initially by the main steam line flow restrictor. The flow from the downstream side of the break is limited initially by the downstream break area. The decrease in steam pressure at the turbine inlet initiates closure of the main steam isolation valves within about 200 milliseconds after the break occurs (see Subsection 7.3 "Primary Containment Isolation System"). Also, main steam isolation valve closure signals are generated as the differential pressures across the main steam line flow restrictors increase above isolation setpoints. The instruments sensing flow restrictor differential pressures generate isolation signals within about 500 milliseconds after the break occurs.

A reactor scram is initiated as the main steam isolation valves begin to close (see Subsection 7.2, "Reactor Protection System"). In addition to the scram initiated from main steam isolation valve closure, voids generated in the moderator during depressurization contribute significant negative reactivity to the core even before the scram is complete. Because the main steam line flow restrictors are sized for the main steam line break accident, reactor vessel water level remains above the top of the fuel throughout the transient.

14.11.5.1.3 Coolant Loss and Reactor Vessel Water Level

The steam flow rate through the downstream side of the break increases from the initial value of 1000 lb/sec in the line to 2000 lb/sec (about 200 percent of rated flow for one steam line) with critical flow initially occurring at the flow restrictor. The steam flow rate was calculated using an ideal nozzle model. Tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions have been used to substantiate the flow models capability to predict the steam flow behavior in the presence of a flow restrictor.

The steam flow rate through the downstream side of the break consists of equal flow components from each of the unbroken lines. The pipe resistance and local restrictions in the unbroken lines result in critical flow initially occurring at the downstream side break location. The steam flow rate in each of the unbroken lines increases from an initial value of 1000 lb/sec to 1530 lb/sec.

The total steam flow rate leaving the vessel is approximately 6600 lb/sec, which is in excess of the steam generation rate of 4000 lb/sec. The steam flow-steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 35 psi/sec. The formation of bubbles in the reactor vessel water causes a rapid rise in the water level. The analytical model used to calculate level rise predicts a rate of rise of about 6 feet/second. Thus, the water level reaches the vessel steam nozzles at 2 to 3 seconds after the break, as shown in Figure 14.11-16. From that time on a two-phase mixture is discharged from the break. The two-phase flow rates are determined by vessel pressure and mixture enthalpy.[15]

The vessel depressurization is calculated using a digital computer code in which the reactor vessel is modeled as five major nodes. The model includes the flow resistance between nodes, as well as heat addition from the core.

As shown in Figure 14.11-16, two-phase flow is discharged through the break at an almost constant rate until late in the transient. This is the result of not taking credit for the effect of valve closure on flow rate until isolation valves are far enough closed to establish critical flow at the valve locations. The slight decrease in discharge flow rate is caused by depressurization inside the reactor vessel. The linear decrease in discharge flow rate at the end of the transient is the result of the assumption regarding the effect of valve closure on flow rate after critical flow is established at the valve location.

The following total masses of steam and liquid are discharged through the break prior to isolation valve closure:

[15] Moody, F. J.: "Two Phase Vessel Blowdown From Pipes", Journal of Heat Transfer, ASME Vol, 88, August 1966, page 285.

Steam	25,000 pounds (19,874 pounds for dose evaluation)
Liquid	160,000 pounds (43,740 pounds for dose evaluation)

Analysis of fuel conditions reveals that no fuel rod perforations due to high temperature occur during the depressurization, even with the conservative assumptions regarding the operation of the recirculation and feedwater systems. MCHFR remains above 1.0 at all times during the transient. MCHFR has been replaced by a similar fuel thermal parameter called MCPR (Minimum Critical Power Ratio). No fuel rod failures due to mechanical loading during the depressurization occur because the differential pressures resulting from the transient do not exceed the designed mechanical strength of the core assembly.

After the main steam isolation valves close, depressurization stops and natural convection is established through the reactor core. No fuel cladding perforation occurs even if the stored thermal energy in the fuel were simply redistributed while natural convection is being established; cladding temperature would be about 1000°F, well below the temperatures at which cladding can fail. Thus, it is concluded that even for a 10.5 second main steam isolation valve closure, fuel rod perforations due to high temperature do not occur. For shorter valve closure times, the accident is less severe. After the main steam isolation valves are closed, the reactor can be cooled by operation of any of the normal or standby cooling systems. The core flow and MCHFR during the first 10.5 seconds of the accident are shown in Figures 14.11-17 and 14.11-l8. Since the MCHFR never drops below 1.0, the core is always cooled by very effective nucleate boiling. Transient limits for nonstandard test or demonstration fuel bundles are given in Appendix N.

14.11.5.2 Radioactive Material Release

14.11.5.2.1 Assumptions

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the nuclear system process barrier outside the secondary containment:

a. The amounts of steam and liquid discharged are as calculated from the analysis of the nuclear system transient.

b. The concentrations of biologically significant radionuclides contained in the coolant discharged as liquid (which subsequently flashes to steam) and the coolant discharged as steam are based on the ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors" methodology. The halogens considered are I-131, I-132, I-133, I-134, I-135.

The values obtained by the ANSI/ANS-18.1 evaluation are then scaled to represent a dose equivalent I-131 concentration of 32 μCi/cc which is 10 times the equilibrium value for continued full power operation allowed by Technical Specifications. Since this value is 10 times the equilibrium value for continued full power operation allowed by Technical Specifications and several orders of magnitude higher than normal reactor coolant concentrations, considerable conservatism is included in the analysis.

c. The concentration of noble gases leaving the reactor vessel at the time of the accident are based on the ANSI/ANS-18.1 concentrations with an appropriate scaling based on NEDO-10871, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms".

d. It is assumed that the main steam isolation valves are fully closed at 5.5 seconds after the pipe break occurs. This allows 500 milliseconds for the generation of the automatic isolation signal and 5 seconds for the valves to close. The valves and valve control circuitry are designed to provide main steam line isolation in no more than 5.5 seconds. The actual closure time setting for the isolation valves is less than 5 seconds.

e. Due to the short half-life of nitrogen-16 the radiological effects from this isotope are of no major concern and are not considered in the analysis.

14.11.5.2.2 Fission Product Release From Break

Using the above assumptions, the following amounts of radioactive materials are released from the nuclear system process barrier:

Noble gases	1.5×10^1 Ci
Iodine 131	1.3×10^2 Ci
Iodine 132	1.1×10^3 Ci
Iodine 133	8.6×10^2 Ci
Iodine 134	1.8×10^3 Ci
Iodine 135	1.2×10^3 Ci

The above releases take into account the total amount of liquid released as well as the liquid converted to steam during the accident.

14.11.5.2.3 Steam Cloud Movement

The following initial conditions and assumptions are used in calculating the movement of the steam cloud:

a. Additional flashing to steam of the liquid exiting from the steam line break will occur due to its superheated condition in the atmosphere.

b. The pressure buildup inside the turbine building will cause the blowout panels to function, resulting in release of the steam cloud in a matter of seconds.

c. Steam cloud rise as predicted by the following equation could vary between 100 and 600 meters depending upon the assumptions made regarding wind speed.[16]

$$h = \frac{11Q^{1/3}}{u}$$

where:

h = Height of cloud rise (ft)
u = Wind speed (ft/sec)
Q = Heat output of cloud (cal/sec)

While the effect of cloud rise is a physical reality, this effect has been neglected for this accident and the assumption is made that the steam cloud does not attain an elevation greater than the height of the turbine building.

The following assumptions and initial conditions are used in calculating the radiological effects of the steam line break accident:

a. The steam cloud movement parameters of paragraph 14.11.5.2.3, and

b. All of the activity released from the reactor vessel to the turbine building is conservatively assumed to escape to the environment.

14.11.5.3 Radiological Effects

The resulting radiological exposures are shown in Table 14.11-11. These values are well within the guideline doses set forth in 10 CFR 100.

Since all of the activity is released to the environment in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated.

It is concluded that no danger to the health and safety of the public results as a consequence of this accident.

[16] Singer, I. A., Frizzola, J. A., Smith M . E., "The Predition of the Rise Of A Hot Cloud From Field Experiments, "Journal of the Air Pollution Control Association, November, 1964.