14.6 ANALYSIS OF DESIGN BASIS ACCIDENTS - UPRATED

This section contains general descriptions of the evaluation of design basis accidents for BFN Units 1, 2, and 3 at uprated conditions. The similar results at pre-uprated conditions can be found in Section 14.11.

14.6.1 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 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.

14.6.2 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 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. A bounding generic evaluation[1] of the CRDA for all BWRs and fuel designs has been performed by GE for plants utilizing the BPWS. For GE analyzed reload cycles in which the BPWS is utilized, a cycle specific CRDA analysis is not required. For GE analyzed reload cycles, the cycle specific CRDA results or a commitment to employ BPWS are contained in the Reload Licensing Report. For AREVA analyzed reload cores, the cycle specific CRDA results are provided in the Reload Licensing Analysis Report.

The BPWS is an improvement over previous group notch sequences with regards to reducing maximum incremental control rod worths. It virtually eliminates the CRDA as an accident of any concern not because it eliminates the possibility of a rod drop occurring, but because the BPWS maintains incremental rod worths to such low values.[2, 3]

The BPWS is effective on a generic basis for all production line reactors and all fuel designs currently in use for initial, reload, and equilibrium core designs.

14.6.2.1 Excursion Analysis Assumptions for GE Analyzed Reload Cores

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

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[1] NEDE 24011-P-A, GESTAR II

[2] NEDO 10527 including Supplements 1 and 2, Rod Drop Accident Analysis for Large BWRs, March 1972

[3] NEDO 21231, Bank Position Withdrawal Sequence, January 1977

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[4] an engineered safeguard, limits the rod drop velocity to less than this value.

b. No credit is taken for the IRM or 15% APRM scram signals. 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.

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.

e. Scram times for the control rods is conservatively assumed to be equal to or greater than the Technical Specification limits. The scram rates which were used in this analysis are tabulated below.

Percent of Rod Insertion	Time (second)
5%	0.475
20%	1.10
50%	2.0
90%	5.0

f. The rod drop accident was evaluated at the time in the fuel cycle at which the consequences are worst.

14.6.2.2 CRDA Analysis and Results for AREVA Licensed Reload Cores

The AREVA analytical methods, assumptions, and conditions for evaluating the excursion aspects of the control rod drop accident have been reviewed and approved by the NRC. AREVA has performed and submitted a generic analysis that correlates deposited enthalpy from a postulated CRDA to steady state parameters calculated on a cycle specific basis. Analyses are performed assuming BPWS rules or equivalent are in force to limit dropped rod worths to reasonable values. The

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

AREVA cycle specific application of the generic CRDA methodology shows that peak deposited enthalpies do not exceed 280 cal/g. For AREVA methods, the most limiting condition to experience a CRDA occurs in the hot standby state. The reload fuel vendors' CRDA methodology conservatively assumes an adiabatic boundary condition at the pellet-gap interface and no direct moderator heating. This prevents heat transfer from the fuel rod to the coolant, thus the deposited enthalpy is equivalent to the energy produced in the fuel. Doppler feedback limits the excursion until the rods are fully inserted.

The core at the time of rod drop accident is assumed to contain no xenon, to be in a hot-startup condition, and to have the control rods in a sequence consistent with BPWS rules or equivalent. For conservatism, eight rods are assumed to be inoperable and remain in the fully inserted position. The location of the inoperable rods are chosen to conservatively increase the worth of the dropped rod. Since the maximum incremental rod worth is maintained at very low values (by BPWS rules or equivalent), the postulated CRDA does not result in peak enthalpies in excess of 280 calories per gram.

The radiological evaluations are based on the assumed failure of 850 fuel rods of a GE fuel type which bound the radiological releases for all fuel rod types in the current core. In the AREVA analysis, rods with deposited enthalpies exceeding 170 cal/g are assumed to fail. If the number of rods exceeding the failure threshold is shown to be below 850, it is concluded that the current radiological evaluation remains applicable.

The results of the peak enthalpy calculation for the current reload cycle are presented in the Reload Licensing Analysis Report. These results demonstrate that the maximum incremental rod worth is below the worth required to result in a CRDA which would exceed 280 cal/g peak fuel enthalpy and that the fuel failures predicted (if any) are fewer than those assumed in the radiological evaluation of record. The conclusion is that the 280 cal/g threshold is protected and that the radiological evaluation accounting for 850 failed fuel rods remains applicable for AREVA fuel.

14.6.2.3 Fuel Damage

Fuel rod damage estimates were initially based upon the UO_2 vapor pressure data of Ackerman[5] 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/g. Two especially applicable sets of data come from the PULSTAR[6]

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[5] 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.

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

and ANL-TREAT[7] 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/g with a minimum period of 2.83 milliseconds. 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.

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

	Test 1	Test 2
Input Energy (cal/g)	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/g is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/g range, and a minimum of 425 cal/g is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

14.6.2.4 BPWS Analysis for GE Analyzed Reload Cores

The accident is analyzed for both the startup range and the power range. The cold startup state will refer to a critical reactor with fuel and moderator temperatures of 20°C, a reactor pressure of one atmosphere, and an initial power fraction of 10^{-8} of rated power level. The hot startup conditions will be defined as a critical reactor at operating pressure, saturated temperature, and initial power fractions of 10^{-6} of rated. Hot standby will be used to define a reactor which is producing sufficient power to maintain all electrical systems without the aid of auxiliary power. This is usually in the 5 to 10% power range. From these definitions, it is obvious that the cold startup and hot startup states will be in the startup range; and that the hot standby case will be in the power range.

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

For the generic BPWS analysis, the fuel designs considered included a single enrichment design with uniform axial gadolinium (Type 1 fuel), a single enrichment design with axially distributed gadolinium (Type 2 fuel), and a mixed enriched, three radial region design (Type 3 fuel). Then the incremental control rod worths were calculated for the Type 1, Type 2, and Type 3 fuel designs for 368, 560, and 748 bundles size cores. These size cores were utilized to represent cores of the general small, medium and large sizes. The highest incremental control rod worth encountered for any of these fuel designs and core sizes was calculated as the beginning of the equilibrium cycle with Type 3 fuel in a 748 bundle size core. This incremental reactivity worth was 0.0083 Δk.

A design basis control rod drop accident with a control rod worth of 0.0083 Δk would result in a peak fuel enthalpy of 135 Cal/g. Since the calculated incremental control rod worth for all other conditions analyzed is less than 0.0083Δk, it follows that the resultant peak full enthalpy due to a design basis control rod accident within the constraints of the BPWS will be less than or equal to 135 Cal/g which is less than both the 170 cal/g and 280 cal/g criteria discussed above.

14.6.2.5 Fission Product Release From Fuel

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

a. Eight hundred fifty fuel rods fail, per General Electric (GE) Licensing Topical Report, NEDO-31400A.

b. The reactor has been operating at design power (with a 1.02 uncertainty factor) with an average fuel burn-up of 35 to 39 GWd/MT prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. The rods that have failed are assumed to have operated at a power peaking factor of 1.5[8].

c. Of the rods that fail, 0.77% of the fuel melts, per NEDO-31400A. The following percentages of radioactive material are released to the reactor coolant from the failed fuel rods[8]:

[8] Regulatory Guide 1.183 and NUREG-0800, Section 15.4.9.

Radionuclide Group	Non-Melted Rods	Melted Rods
Noble Gases	10%	100%
Iodine	10%	50%
Other Halogens	5%	30%
Alkali Metals	12%	25%
Tellerium Group	0%	5%
Barium, Strontium	0%	2%
Noble Metals	0%	0.25%
Cerium Group	0%	0.05%
Lanthanum Group	0%	0.02%

14.6.2.6 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. Of the radioactive material released from the fuel, 100% of the noble gases, 10% of the iodines, and 1% of the remaining radionuclides are assumed to reach the turbines and condensers[8].

b. Activity is assumed to be released from core instantaneously to the condenser.

14.6.2.7 Fission Product Release to Environs

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

a. On reaching the condenser, 100% of noble gases, 10% of iodines, and 1% of the particulate radionuclides are available for release to the environment. Radioactive decay during holdup in the low pressure turbine and condenser is assumed.

b. The accident is assumed to occur while condenser vacuum is being maintained with the mechanical vacuum pump (MVP). 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 beyond the condenser.

c. All of the noble gas activity transferred to the condenser is assumed to be airborne in the condenser. The halogen and particulate activity transferred to the condenser experiences the removal effects of the condensate as described above.

d. The rate at which the condenser activity is discharged to the environment is dependent upon the free volume of the turbine and condenser and the discharge rate of the mechanical vacuum pump. The numerical values appropriate to these parameters are 187,000 ft^3 (low pressure turbine volume plus condenser free volume) and 1,850 cfm mechanical vacuum pump discharge rate.

e. A continuous ground level release of 20 cfm occurs at the base of the stack. The 20 cfm leakage mixes within the rooms at the base of the stack (34,560 ft^3, 50% of 69,120 ft^3 because of incomplete mixing).

f. 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 14.6-8. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.

g. The maximum control room X/Q for the top and bottom of the stack releases is used for each time period. The effective X/Q is a factor of two less than the values listed because of the dual air intake configuration of the control bay ventilation (i.e., one intake is not contaminated).

Based upon these conditions, the fission product release rate to the environment is shown in Table 14.6-1.

14.6.2.8 Radiological Effects

The BFN analysis for the CRDA consists of two potential release paths; condenser leakage at 1% per day into the turbine building or through SJAE and offgas system as analyzed by the NEDO-31400A, and the MVP discharge as analyzed in accordance with Regulatory Guide 1.183. The “worst-case” radiological exposure resulting from the activity discharged from a CRDA and a Regulatory Guide 1.183 source term would be from the MVP release path. The resulting control room dose is less than the 10 CFR 50.67 limit of 5 Rem TEDE. The EAB and LPZ doses from the MVP are well below the Regulatory Guide 1.183 reference values of 6.3 REM TEDE.

The dominant contributor to dose for the CRDA is Iodine 131 (I-131). Table 14.6-1 shows the I-131 activity in four locations (main condenser, stack room, control room, and environment) for the full 30 days of the dose calculation described above. This is an output of the RADTRAD computer code (NUREG/CR-6604) used for the CRDA dose analysis. Radioactive decay is considered in all locations except the environment (i.e., the environment represents a summation of all activity released). The environmental release totals approximately 10 percent of the activity initially reaching the main condenser. The main condenser is depleted of 95% of the activity by about five hours. This is consistent with an 1850 cfm exhaust rate and a 187,000 ft^3 volume (i.e., a release rate of about 0.6 volumes per hour).

14.6.3 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.

ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. For peak cladding temperatures and limiting break sizes for GE and AREVA fuels, see Section 6.5.3.1.

Information on GE LOCA models currently in use is given in NEDO-20566[9] and NEDC-32484P[10]. LOCA models used for AREVA reload fuel analyses are

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

10 General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P, Rev. 6.

described in EMF-2361(P)(A)[12], ANP-3015P[13], and ANP-3152P[14]. 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 NEDC-32484P and NEDO-10320[11].

14.6.3.1 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 Engineered Safeguards.

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.

d. One active single failure within the plant is postulated to occur concurrent with the pipe break.

e. A seismic event is neither postulated to occur concurrently with the LOCA nor as a initiator of the pipe break.

14.6.3.2 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, and the heatup of the fuel.

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11 The General Electric Pressure Suppression Containment Analytical Model, NEDO-10320.

12 EMP-2361(P)(A), Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012.

13 ANP-3015, Revision 0, Browns Ferry Nuclear Plant Units 1, 2, and 3 LOCA Break Spectrum Analyses, AREVA NP, Inc., September 2011.

14 ANP-3152(P), Revision 0, Browns Ferry Nuclear Plant Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel, AREVA NP, Inc., October 2012.

14.6.3.3 Primary Containment Response

BFN Units 1, 2, and 3 use the Mark I primary containment design. The main function of the Mark I containment design is to accommodate pressure and temperature conditions within the drywell resulting from a LOCA or a reactor blowdown through the MSRV discharge piping and, thereby, to limit the release of fission products to values which will ensure off-site dose rates below the 10 CFR 50.67 limits. In the event of a pipe break in the drywell, water and/or steam from the reactor pressure vessel (RPV) are discharged into the drywell. The resulting increase in the drywell pressure forces the water and steam, along with non-condensable gases initially existing in the drywell, through the vents which connect the drywell to the suppression pool. During a reactor blowdown through the SRVs, the steam is directly discharged into the suppression pool. The reactor blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the fluid inventory in the RPV.

The long-term heatup of the suppression pool following a LOCA is governed by the capability of the Residual Heat Removal (RHR) System to remove decay heat which is transferred from the RPV to the suppression pool.

The Primary Containment System requirements are:

Design Pressure	56 psig
Design Temperature	281°F

Minimum containment overpressure following a LOCA and its affect on NPSH for Low Pressure Core Spray (LPCS) and RHR pumps is discussed in Chapter 6.5.5.

14.6.3.3.1 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 14.6.3.1.)

a. The reactor is assumed to be initially operating at the conditions specified in Table 14.6-3. Tables 14.6-4 and 14.6-5 provide additional conditions that apply for the short term containment response and long term containment response, respectively.

b. 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.

c. The sensible heat released in cooling the fuel to the normal primary system operating saturation temperature and the core decay heat were included in the reactor vessel depressurization calculation. Initial high vessel pressure increases the calculated flow rates out of the break; this is conservative for containment analysis purposes.

d. 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 for the outboard valves and 2 minutes for the inboard valves. 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.

e. For the short term containment response analysis, the feedwater flow is assumed to coast down to zero at four seconds into the event. 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.

f. For the long term containment response analysis, the reactor feedwater flow into the reactor continues until all the high energy feedwater (water that would contribute to heating the pool) is injected into the vessel.

g. 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.

   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.

h. The limiting core/containment cooling configuration assumed is the availability of one reactor core spray loop and one RHR loop consisting of two RHR pumps and associated heat exchangers and two associated RHR service water pumps.

i. For the long term containment response analysis, LPCI and LPCS are used to cool the core for the first 600 seconds. After 600 seconds, it is assumed that containment cooling is manually initiated using containment spray.

14.6.3.3.2 Containment Response

The containment performance for the DBA-LOCA response is typically divided into two phases: the short-term initial blowdown period (approximately 30 seconds following a LOCA) and the long-term period which includes the time period after the containment cooling system starts. The short-term containment response determines the peak drywell pressure and the peak drywell LOCA temperature. The long-term containment response determines the peak wetwell (suppression pool) temperature and pressure.

The following subsections provide a description of the dynamics of the containment response during a LOCA along with the calculational methods and results of the short term and long term containment response at power uprated conditions.

14.6.3.3.2.1 LOCA Dynamics

Following the initiation of the LOCA, the primary coolant from the reactor vessel is discharged into the drywell. Most of the noncondensible gases are forced into the suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the suppression chamber via the vacuum breaker system as the drywell pressure decreases due to steam condensation. The LPCS removes decay heat and stored heat from the core, thereby controlling core heatup. 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.

14.6.3.3.2.2 Short-Term Response

The short-term containment pressure and temperature response was re-analyzed at power uprate conditions in accordance with Regulatory Guide 1.49[15] and NEDO-31897[16], using the GE proprietary computer code M3CPT05V. The modeling used in M3CPT is described in NEDO-10320[17], NEDO-20533[18], and NEDE-20566-P-A[19]. The short-term containment response is controlled by the reactor blowdown during the LOCA. The reactor blowdown rate is dependent on the reactor initial thermal hydraulics conditions, such as vessel dome pressure and the mass and energy of the fluid inventory in the RPV. However, the reactor blowdown is relatively insensitive to the initial reactor power.

The M3CPT analyses were performed using blowdown flow rates based on the GE code LAMB08 blowdown model[20]. In using the LAMB blowdown model, the blowdown flow rates were calculated first. The LAMB flow rates were then used as input to M3CPT.

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15 Regulatory Guide 1.19

16 GE Nuclear Energy, “Generic Guidelines for GE Boiling Water Reactor Power Uprate,” Licensing topical Report NEDO-31897, Class I (Non-proprietary), February 1992, and NEDC-31897P-A, Class III (Proprietary), May 1992

17 NEDO-10320, “The GE Pressure Suppression Containment Analytical Model,” April 1971

18 NEDO-20533, “The General El3ectric Mark III Pressure Suppression Containment System Analytical Model,” June 1974

19 NEDO-21052, “Maximum Discharge of Liquid-Vapor from Vessels,” September 1975

20 NEDE-20566-P-A, General Electric Model for LOCA Analysis in Accordance with 10CFR50 Appendix K,” September 1986

The following four reactor operating points on the power/flow map were selected for evaluation to envelope the full range of reactor operating conditions:

Case 1 -	102% of uprated power, 100% core flow with normal feedwater temperature.
Case 2 -	102% of uprated power, 100% core flow with feedwater temperature reduction.
Case 3 -	102% of uprated power, 81% core flow with feedwater temperature reduction [MELLLA point].
Case 4 -	63% of uprated power, 38% core flow with feedwater temperature reduction [natural circulation line-MELLLA rod line intersection].

The containment response for the Increased Core Flow (ICF) state point was not analyzed since it is bounded by the containment response for the above power/flow state points.

Table 14.6-6 presents the results of all the power/flow state points analyzed. The results demonstrate that the maximum drywell pressure and maximum differential pressure between the drywell and wetwell during operation at uprated power remain within the containment design limits.

The peak drywell pressures for all points analyzed are well below the design limit. The highest peak short-term drywell pressure and temperature for power uprate conditions (50.6 psig, 297°F) occur at the MELLLA point (Case 3) for Unit 2 and Unit 3. The peak drywell pressure and the peak drywell gas temperature for Unit 1 are 48.5 psig and 295.2°F, respectively. Although the calculated peak drywell atmosphere temperature is higher than the drywell shell design value of 281°F, the shell temperature will not exceed 281°F. This is because drywell atmosphere temperature exceeds 281°F for a short duration following the blowdown, and it would take a longer time for the drywell shell to heat up to 281°F. Thus, the drywell shell is expected to remain below the design temperature of 281°F. Additionally, the safety components in the drywell that must function following a LOCA have been successfully tested in a steam atmosphere at higher temperatures than the containment design temperature of 281°F (FSAR Section 12.2.2.7.3).

Plots showing the limiting DBA-LOCA short-term temperature and pressure response in the drywell and wetwell at power uprate conditions are given in Figures 14.6-1 and 14.6-2, respectively.

14.6.3.3.2.3 Long-Term Response

As the operating power level is increased due to power uprate, the decay power increases and the long-term pressure suppression pool temperature will potentially increase. The most limiting DBA-LOCA case with respect to peak pressure suppression pool temperature, a double-ended recirculation suction line break, was analyzed at power uprate conditions using the SHEX-04V code[21]. In the long-term response evaluation at power uprate conditions, the ANSI/ANS 5.1 - 1979 decay heat model plus $2_{x}\rho$ uncertainty was used.

The results of the analysis shows the peak pressure suppression pool temperature is less than 177°F for 105% power uprate. Figures 14.6-3 and 14.6-4 show the long term wetwell and drywell temperature response on Units 2 and 3. Figure 14.6-5 provides the long term pressure response of the drywell and wetwell on Units 2 and 3. The same case was re-analyzed at the pre-uprate power conditions to assess the impact of power uprate on peak pool temperature on a common analysis basis. The comparison indicates that power uprate increases the peak suppression pool temperature by 2°F. For Unit 1 the peak suppression pool temperature is 187.3°F, which is based on a 120% power uprate analysis.

For Units 2 and 3, the unlikely occurrence that the RHR service water temperature exceeds the design value of 92°F, an allowable derated operating power map has been developed to enable the operator to determine the maximum allowed operating power limit for a range of service water temperatures. This power map is included in the Technical Specification for the ultimate heat sink. The limit assumes the plant power level has been within the limit for a long enough period of time such that it can be considered a steady state condition. This assumption is required because during a power reduction the total decay heat lags the instantaneous power level. Based on historical operating plant data, 95°F is chosen as the upper bound for the RHR service water temperature range. A long-term containment sensitivity study was performed to identify the maximum acceptable core thermal power as a function of RHR service water temperature in order to maintain the peak pressure suppression pool temperature at 177°F and, thus, satisfy the temperature limit per the Torus Integrity Long-Term Program Plant Unique Analysis Report[22].

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21 Letter to Patrick W. Marriot (GE) from William T. Russel (NRC) forwarding the Staff Position Power on General Electric Boiling Water Reactor Power Uprate Program (TAC No. M79384), September 30, 1991

22 Report CEB-83-34 R2, “Browns Ferry Nuclear Plant Torus Integrity Long-Term Program Plant Unique Analysis Report (PUAR):

The Unit 1 analysis is based on a RHR service water temperature of 95°F and the Unit 1 Technical Specifications do not include a derated power operating map. On Unit 1, the peak suppression pool temperature is 187.3°F[23].

14.6.3.3.3 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 long term 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 containment pressure response curves presented in Section 14.6.3.3.2 do not reflect the negligible long term pressure increase due to this phenomena.

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 extent of the metal-water reaction 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

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23 Report GE-NE-0000-0011-4656, "Browns Ferry Unit 1 Asset Enhancement Program Containment System Response"

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 14.6-6. 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 "non-steaming" 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 second,

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.

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 14.6-6 present the results of the parametric investigation.

14.6.3.4 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 the ORIGEN computer code with a 1.02 multiplier per Regulatory Guide 1.183.

b. The reactor has been operating at design power (3952 MWt) for a 24 month fuel cycle. The average fuel burnup is 35 to 39 GWd/MT prior to the accident.

c. The radionuclides considered include those identified as being potentially important contributors to TEDE in NUREG/CR-6604.

d. The core inventory release fractions, timing, and chemical form are those specified in Regulatory Guide 1.183. Table 14.6-7 gives the bounding core inventory of each isotope .

14.6.3.5 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. Primary containment atmosphere is released via main steam isolation valve leakage to the high and low pressure turbines and the condenser. Primary containment atmosphere is released directly to the Standby Gas Treatment System during operation of the Containment Atmospheric Dilution (CAD) System. Primary containment atmosphere is released above the Units 1 and 2 Reactor Buildings via leakage of the Unit 1 and 2 hardened containment venting system isolation valves. Primary containment atmosphere is released to the top of the stack via leakage of the Unit 3 hardened wetwell vent isolation valves. The Emergency Core Cooling Systems (ECCS) leak into the secondary containment. The following assumptions were used in calculating the amounts of fission products released from the primary containment:

a. The primary containment minimum free volume (drywell and wetwell) is 278,400 ft^3. The drywell volume is 159,000 ft^3 and the torus gas space volume is 119,400 ft^3. The drywell torus gas space volumes are treated as separate volumes until after the activity release to the containment is complete and then these volumes are assumed to be well mixed. The activity release is entirely to the drywell.

b. The primary to secondary containment leak rate was taken as two percent volume per day (232 cfh).

c. The four main steam lines are assumed to leak a total of 150 scfh which is the Technical Specification limit.

d. The containment vent system flow path 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. The Unit 3 hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack. The Unit 1 and 2 hardened containment vent isolation valves leak a total of 10 scfh to the independent release points above the Unit 1 and 2 Reactor Buildings. Release associated with leakage from the Unit 1 and 2 hardened containment vent isolation valves is assumed to begin at 11 hours.

f. Twenty gpm ECCS leakage into secondary containment in accordance with NUREG-0800, Section 15.6.5, Appendix B.

g. No credit is taken for spray removal in the containment.

h. Natural removal rates for particulates in the drywell are based on the correlations of NUREG-CR-6604. For elemental iodine, the natural removal coefficients for removal of plateout are based on the expressions of SRP 6.5.2.

i. For the purpose of suppression pool pH control, the accident is assumed to be a recirculation line break.

Additionally, an analysis evaluated the suppression pool pH in the event of a DBA LOCA involving fuel damage. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0; thus, ensuring that the particulate iodine (Cesium Iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine.

The calculation methodology was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Specifically, credit was taken for sodium pentaborate solution addition to the suppression pool water as a result of SLCS operation.

The initial effects on suppression pool pH come from rapid fission product transport and formation of cesium compound, which would result in increasing the suppression pool pH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the suppression pool over the first days of the event, the suppression pool water would become more acidic. The buffering effect of SLCS injection within several hours is sufficient to offset the effects of these acids that are transported to the pool. Sufficient sodium pentaborate solution is available to maintain the suppression pool pH at or above 7.0 for 30 days post-accident.

14.6.3.6 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 primary containment atmosphere leakage to secondary containment mixes instantaneously and uniformly within the secondary containment.

b. The effective mixing volume of the secondary containment is 1,311,209 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. However, negative pressure would be re-established in secondary containment prior to fission product release times specified by Regulatory Guide 1.183. 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 24,750 cfm), all releases to the environment from secondary containment are through the SGTS filters via the plant stack. The case with three trains in operation is the limiting condition.

d. The containment vent system flow path 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. The ECCS systems leak reactor coolant directly to the secondary containment. The maximum water temperature is less than 212°F. The volume available for mixing is 1.31E5 ft^3. Ten percent of the iodine in the ECCS leakage is assumed to become airborne.

f. Filter efficiency for the SGTS was taken as 90 percent for organic and 0% inorganic (elemental) iodine.

g. Release to the environment from the plant stack is composed of three flow paths. A continuous ground level release of 20 cfm occurs at the base of the stack. This flow results from SGTS leakage through the backdraft dampers in the base of the stack. Subsection 5.3.3, "Secondary Containment System Description" describes the backdraft dampers. The 20 cfm leakage mixes uniformly within the rooms at the base of the stack (50% of the room volume of 69,120 ft^3). The remaining SGTS flow exits the stack at a height of 183 meters above ground elevation. The Unit 3 hardened wetwell vent isolation valves leak a total of 10 scfh to the top of the offgas stack. Releases associated with the hardened wetwell vent isolation valves for Unit 3 are bounded by releases from the Unit 1 hardened containment vent isolation valves. The hardened wetwell vent isolation valve leakage enters the stack above the divider deck and exits the top of the stack.

h. Fumigation conditions exist for 30 minutes when the post-accident control room accumulated dose rate is the maximum.

i. 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 14.6-8. Control room X/Q values for the base of the stack releases are calculated using the computer code ARCON96. For sites, such as BFN, with control room ventilation intakes that are close to the base of tall stacks, ARCON96 underpredicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes are evaluated using the methods of Regulatory Guides 1.145 and 1.111.

j. The maximum control room X/Q for the top and bottom of the stack releases is used for each time period. Note that the effective X/Q is a factor of two less than the values listed because of the dual air intake configuration of the control bay ventilation.

k. The Unit 1 and 2 hardened containment vent isolation valves leak a total of 10 scfh to the independent release points above the Unit 1 and 2 Reactor Buildings with a delay of 11 hours for leakage to reach the release point. A bounding control room X/Q is used for each time period for this release path.

Main Steam Isolation Valve Leakage Releases

The leakage from primary containment via the MSIVs is transferred 1) to the main turbine (high pressure and low pressure) via the four steam lines and 2) to the condenser via the alternate leakage treatment (ALT) flow path formed by the steam line drain. The leakage from the turbine and condenser migrates to the turbine deck and subsequently is exhausted to the atmosphere via the turbine building roof vents with no credit for hold-up or removal in the Turbine Building. The path takes advantage of the large volume of the main steam lines and the condenser to hold up and plate out fission products in the MSIV leakage effluent. The following assumptions were used to calculate the fission product activity released to the environment from the turbine building:

a. The four main steam lines are assumed to leak a total of 150 scfh which is the Technical Specification limit. The direct leakage path to the turbines processes only 0.5% of the total leakage. The remainder goes to the condenser via the ALT flow path. The main steam piping from the outermost isolation valve up to the turbine stop valve, the bypass/drain piping to the main condenser and the main condenser will retain their structural integrity during and following a safe-shutdown earthquake (SSE).

b. Aerosol and elemental iodine removal due to sedimentation is credited in the main steam lines and in the main condenser. Aerosol settling velocities for sedimentation are determined for the steam lines and the main condenser per the AEB 98-03 distribution. Settling velocities are based on removal coefficients for the different volumes considering prior volume sedimentation removal. Elemental iodine removal in the steam lines utilizes the Bixler model of NUREG/CR-6604. The elemental iodine removal rate in the condenser is conservatively assumed to be the same as that for particulate.

c. The free volume of the low pressure turbines is 51,000 ft^3 and the effective volume of the condenser is 122,400 ft^3 (90% of the total condenser volume).

d. No credit is taken for holdup in the turbine building.

e. Ground level atmospheric dispersion coefficients, X/Q, for releases from the turbine building roof exhaust applicable to the time periods, distances, and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values are calculated using the computer code ARCON96.

14.6.3.7 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 maximum 2-hour TEDE for the exclusion area boundary (EAB) (1,465 meters), and the corresponding 30-day TEDE at the low population zone (LPZ) boundary (3,200 meters).

The offsite doses are calculated using the RADTRAD code (NUREG/CR-6604). RADTRAD is a radiological consequence analysis code used to model plan control volumes for radionuclide transport and removal and account for atmospheric dispersion of offsite and control room locations by use of appropriate X/Qs.

The largest calculated total offsite dose is well within the 10 CFR 50.67 limit.

Control Room

The control room doses are calculated using RADTRAD (NUREG/CR-6604). The model accounts for the atmospheric dispersion to the dual control room intakes by use of appropriate X/Qs and models the control bay habitability zone with no credit taken for the Control Room Emergency Ventilation System (CREVS) filters (i.e., 6717 cfm of unfiltered inleakage into the Control Room), occupancy times, breathing rates in accordance with Regulatory Guide 1.183 and calculates the TEDE. 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 model accounts for the control room geometry (210,000 ft^3).

The direct gamma dose contribution from the piping inside secondary containment and the secondary containment atmosphere 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 suppression pool water in the event of a LOCA.

All of these exposure mechanisms (unfiltered pressurization flow, unfiltered inleakage, and direct dose) are combined to produce a total control room dose for the duration of the accident. Since CREVS has dual air intakes placed on opposite sides of the control building and can function with a single active failure in the inlet isolation system, in accordance with NUREG-0800, the control room dose is divided by a factor of 2 to account for dilution effects. The 30 day integrated post-accident doses in the control room are within the limits of 5 REM TEDE as specified in 10 CFR 50.67.

14.6.4 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 Subsections 14.6.4.1 and 14.6.4.2 provides the analyses for the dropping of a 7 x 7 assembly and 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. Evaluations of other fuel types have been performed as a comparison of the fuel damage to the 7x7 fuel design. The activity release for these other fuel types is bounded by the GE 7x7 case. The historical and current calculated doses are much less than the regulatory guidelines.

The refueling accident results documented in this section are applicable for fuel cycles containing an initial reload of new AREVA fuel, including the use of blended, low-enriched uranium (BLEU). The AREVA fuel load chain is different from GE assembly designs because the load is distributed through the center water channel rather than through the rods.

However, the failure mechanisms for the AREVA assemblies will produce similar number of rod failures as in the GE14 design.

14.6.4.1 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).

14.6.4.2 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 17,000 ft-lb for a 7 x 7 fuel bundle and approximately 18,150 ft-lb for a 8 x 8 fuel bundle. This energy 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

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

where 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 17,000 or 13,600 ft-lbs of energy for a 7 x 7 fuel bundle and 0.80 x 18,150 or 14,500 ft-lbs of energy for a 8 x 8 fuel bundle. 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 49 (7 x 7 fuel bundle) or 62 (8 x 8 fuel bundle) 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 41 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 41 x 4 or 41,000 ft-lbs for the 7 x 7 fuel or 250 x 54 x 4 or 54,000 ft-lbs for the 8 x 8 fuel 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:

7 x 7 fuel $$\frac{0.5 \times 13{,}600 \times \left(\frac{11}{11 + 17}\right)}{250} = 11$$

8 x 8 fuel $$\frac{0.5 \times 14{,}500 \times \left(\frac{11}{11 + 17}\right)}{250} = 12$$

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

		7 x 7	8 x 8	
Dropped assembly	-	49	62	rods (bending)
Struck assemblies	-	32	32	tie rods (bending)
Struck assemblies	-	11	12	rods (compression)
		92	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:

$$7 \times 7 \quad \frac{\frac{0.19}{2} \times 17{,}000 \times \frac{11}{11 + 17}}{250} = 3$$

$$8 \times 8 \quad \frac{\frac{0.19}{2} \times 18{,}150 \times \frac{11}{11 + 17}}{250} = 3$$

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 resulting from the accident is as follows:

	7 x 7	8 x 8	
First impact	92	106	rods
Second impact	19	19	rods
Third impact	0	0	rods
	111	125	failed rods (total)

14.6.4.3 Fission Product Release From Fuel

The radiological dose consequences resulting from a refueling accident have been evaluated using Alternative Source Terms (AST) in accordance with 10 CFR 50.67 and the guidelines of Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

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

a. The reactor has been operating at design power (3952 MWt) for 24 month fuel cycle. The average fuel burnup is 35 to 39 GWd/MT prior to the accident. 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 using the ORIGEN code at a core power of 4031 MWt modified with a power peaking factor of 1.5 and Regulatory Guide 1.183 power factor of 1.02 with a decay of 24 hours.

c. One hundred eleven fuel rods are assumed to fail. This was the conclusion of the analysis of mechanical damage to the fuel based on the GE 7x7 fuel design.

14.6.4.4 Fission Product Release to Secondary Containment

The following assumptions were used to calculate the fission product release to the secondary containment (per Regulatory Guide 1.183):

a. Fraction of Fuel Rod Inventory Released (infinite decontamination for nuclides other than iodine and noble gases):

Noble Gases (Except Kr 85)	5 percent
Kr 85	10 percent
Iodines (Except I-131)	5 percent
I-131	8 percent

b.	Iodine Decontamination Factor in Reactor Cavity Pool Water	200 elemental and organic
c.	Iodine Species	99.85% elemental 0.15% organic

14.6.4.5 Fission Product Release to Environs

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

a. The release is assumed to be an instantaneous ground level release to the environment with no holdup time in secondary containment. Accordingly, no credit is taken for filtering by the standby gas treatment system and no credit is taken for an elevated release at the main stack.

b. No credit is taken for isolation of the control room nor for any filtering by the control room emergency ventilation system.

c. The X/Q for the control room is reduced by 50% to reflect the credit for dual control room air intakes as allowed by Standard Review Plan Section 6.4.

d. Control Room Free Volume - 210,000 ft^3

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 111 fuel pins (assuming GE 7x7 fuel design). The inventory described above will be released from the fractured fuel rods. A decontamination factor of 200 for elemental and organic is applicable for iodine released at depth under water. The radioactive releases to the air space above the pool are released instantaneously to the atmosphere with no holdup in secondary containment and no filtering by the Standby Gas Treatment System. The assumptions used to evaluate the fuel handling design basis accident event are defined in Nuclear Regulatory Commission Regulatory Guide 1.183. Further guidance is contained in the Standard Review Plans in NUREG-800, Section 15.0.1.

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.

The fuel handling accident was evaluated using RADTRAD computer programs as described in Section 14.6.3.7. The X/Q values based on the refueling vents from 0-2 hours were used in computing the dose consequences of this release.

14.6.4.6 Radiological Effects

The radiological exposures following the refueling accident have been evaluated in the control room, at the site boundary, and at the LPZ boundary. The calculated dose assumes that all of the activity is exhausted instantaneously through a roof vent; with no credit for holdup time nor filtering by SGTS.

Boundary dose resulting from design basis accident events has been judged by comparing the dose to the 10 CFR 50.67, “Accident Source Term,” limits. This regulation uses radiation doses of 25 Rem TEDE for doses to the public and 5 Rem TEDE for the control room as guides under accident conditions. In the Standard Review Plan, NUREG-800, the limits for doses to the public are reduced by 25 percent to 6.3 Rem TEDE. The calculated doses are much less than the guidelines (< 6.3 Rem TEDE for EAB and LPZ and < 5 Rem TEDE for the control room).

14.6.5 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 14.6-7 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 control room and offsite persons.

14.6.5.1 Nuclear System Transient Effects

14.6.5.1.1 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 the power associated with maximum mass release.

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

i. 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 with a three second time constant.

14.6.5.1.2 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.

14.6.5.1.3 Coolant Loss and Reactor Vessel Water Level

The mass release during a main steamline break outside containment was analyzed at full power and hot standby conditions. At full power, the initial steam flow rate through the break is approximately 7300 lb/sec, while the steam generation rate is almost 4000 lb/sec. The break flow-steam generation mismatch causes a depressurization of the reactor vessel. 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 4 to 5 seconds after the break.

At hot standby, the initial break flow is almost 6600 lb/sec as shown in Figure 14.6-8; but the steam generation rate is about 27 lb/sec. The rise in reactor water level is much faster and reaches the vessel steam nozzles in about one second after the break. 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.[23] Due to the longer duration of two-phase break flow, the hot standby conditions result in much more liquid flowing through the break than at full power such that the total mass release is about 70% greater at hot standby than at full power.

As shown in Figure 14.6-8, 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 a 5.5 second isolation valve closure:

Steam	11,975 pounds
Liquid	42,215 pounds

The evaluation of fuel performance used a bounding time of 10.5 seconds for closure of the main steam isolation valves. 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.

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23 Moody, F. J,: “Two Phase Vessel Blowdown from Pipes,” Journal of Heat Transfer, ASME Vol, 88, August 1966, page 285.

After the main steam isolation valves close, depressurization stops and natural convection is established through the reactor core. Even if the event is initiated from full power (which has a much lower mass release) with a delayed main isolation valve closure, 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 14.6-9 and 14.6-10. 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.

14.6.5.2 Radioactive Material Release

14.6.5.2.1 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, and

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/gm which is greater than the 26 μCi/gm maximum Technical Specification limit and 10 times the equilibrium value for continued full power operation allowed by Technical Specifications.

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.

f. Atmospheric dispersion coefficients, X/Q, for ground level releases from the turbine building exhaust are used. X/Q values applicable to the time periods, distances and geometric relationships (offsite and control room) are shown in Table 14.6-8. Control room X/Q values are calculated using the computer code ARCON96.

g. All of the activity released from the reactor vessel to the Turbine Building is conservatively assumed to escape to the environment.

14.6.5.2.2 <u>Fission Product Release From Break</u>

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

Noble gases	1.342×10^3 Ci
Iodine 131	5.254×10^1 Ci
Iodine 132	4.737×10^2 Ci
Iodine 133	3.533×10^2 Ci
Iodine 134	8.549×10^2 Ci
Iodine 135	5.031×10^2 Ci

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

14.6.5.3 <u>Radiological Effects</u>

The control room dose is divided by 2 because of the dilution effect of the dual air intake configuration of the control bay ventilation. Shine due to radioisotopes in the Turbine Building is also accounted for in the total control room operator dose. The shine is not divided by 2. The control room operator doses due to a MSLB are less than the 10 CFR 50.67 limit of 5 Rem TEDE. The offsite doses are less than the 10 CFR 50.67 limit of 25 Rem TEDE for the maximum Technical Specification reactor coolant (32 μCi/gm I-131 equivalent). Also, the offsite doses are less than 10% of the 10 CFR 50.67 limits for the maximum equilibrium reactor coolant (3.2 μCi /gm).

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