6.7 Radiological Consequences Evaluations (Doses)

6.7.1 Introduction and Background

The radiological consequences for the following design basis accidents (DBAs) were reanalyzed to support the Power Uprating effort:

- Main steamline break (MSLB)
- Locked reactor coolant pump (RCP) rotor
- Rod ejection
- Steam generator tube rupture (SGTR)
- Large-break loss-of-coolant accident (LBLOCA)
- Waste gas decay tank (GDT) rupture
- Volume control tank (VCT) rupture
- Fuel-handling accident (FHA)

All of these accidents are currently addressed in the licensing submittal (Reference 1) for implementing alternative source term for the Kewaunee Nuclear Power Plant (KNPP), using the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analyses performed for the Uprate Project follow the methodology from the licensing submittal (Reference 1), and were performed at the nominal core power level of 1650 MWt but included margin in key input parameters to provide an indication that sufficient margin exists to allow a large power uprating. The analyses from the licensing submittal analyses have all been updated using input assumptions consistent with the proposed uprated nominal core power of 1772 MWt.

For each accident, the total effective dose equivalent (TEDE) doses are determined at the site boundary (SB) for the limiting 0- to 2-hour period, at the low-population zone (LPZ) boundary for the duration of the accident and in the Control Room for 30 days.

6.7.1.1 Input Parameters and Assumptions

The assumptions and inputs described in this section are common to various analyses discussed in the following sections. These assumptions and input are consistent with those presented in the licensing submittal (Reference 1). Each accident and the specific input assumptions are described in detail in subsections 6.7.2 through 6.7.9.

The dose conversion factors (DCFs) used in determining the committed effective dose equivalent (CEDE) or inhalation dose are from the Environmental Protection Agency (EPA) Federal Guidance Report No. 11 (Reference 3). The TEDE dose is equivalent to the CEDE dose plus the acute dose for the duration of exposure to the cloud. The γ-body (acute) doses are based on the average disintegration energies for the iodine isotopes, and from the ICRP Publication 38 (Reference 4) for the remainder of the nuclides (except the noble gases). The dose conversion factors for the noble gases are taken from ICRP Publication 30 (Reference 5). The nuclide data are all listed in Table 6.7-1.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 6.7-2.

The offsite dose acceptance limits are based on 10CFR50.67 guidance of 25 rem TEDE. Depending on the event, the acceptance limit is 100 percent of 10CFR50.67 or a fraction of these guidelines. Some events are designated as having a dose limit that is 25 percent of that limit (6.3 rem TEDE). Other events are specified with a dose limit that is 10 percent of that limit (2.5 rem TEDE).

Parameters modeled in the Control Room personnel dose calculations are provided in Table 6.7-3. These parameters include normal operation flow rates, emergency operation flow rates, Control Room volume, filter efficiencies, Control Room operator breathing rates, and atmospheric dispersion factors. The Control Room dose acceptance limit from 10CFR50.67 is 5 rem TEDE.

Section 7.6 of this report describes the calculation of the core and coolant activity. The core fission product activity modeled in the radiological consequences analyses for the locked rotor, rod ejection, and large-break loss-of-coolant accident (LBLOCA) is provided in Table 6.7-4, and was calculated modeling the third transition cycle. For the three accidents, a multiplier was developed that would yield the same, or less dose for variations of a core average enrichment of 4.5 w/o ±10 percent, a core mass of 49.1 metric ton unit (MTU) ±10 percent, and a cycle length of 493.6 effective full-power day (EFPD) ±10 percent. For locked rotor and rod ejection, the multiplier is 1.06, and for the LBLOCA, the multiplier is 1.03. The doses calculated using the core activity data in Table 6.7-4 were multiplied by these factors, and the results are reported in the appropriate sections of this report. The nominal reactor coolant activity modeled in the radiological consequences analyses, based on 1 percent fuel defects for noble gases and alkali

metals, and 1.0 μCi/gm dose equivalent I-131 (DE I-131) for iodine, is provided in Table 6.7-5. The data in Table 6.7-5 bounds the variations in core average enrichment, core mass, and cycle length, discussed above.

6.7.1.2 Iodine Spiking Models

A number of accident analyses take iodine spiking into consideration (for example, main steam line break [MSLB], steam generator tube rupture [SGTR]).

For the pre-existing iodine spike, it is assumed that a reactor transient has occurred prior to the accident and has raised the primary coolant iodine concentration to 60 μCi/gm of DE I-131 (this is the Technical Specification limit for transient elevated iodine activity in the primary coolant). For the accident-initiated iodine spike, it is assumed that the reactor trip associated with the accident creates an iodine spike, which increases the iodine release rate from the fuel to the reactor coolant. The spike iodine release rate is a multiple of the maximum equilibrium release rate (where the equilibrium release rate is that rate corresponding to maintaining a primary coolant concentration of 1.0 μCi/gm of DE I-131, which is the maximum concentration allowed by the Technical Specifications for continuous operation). Regulatory Guide 1.183 (Reference 2) requires a spike multiplier of 500 for the steamline break, and allows a multiplier of 335 for SGTR. For this analysis, the SGTR spike factor was conservatively increased from 335 to 500, so that the same iodine appearance rates are used for the two events.

The primary coolant iodine concentrations associated with a pre-existing iodine spike are provided in Table 6.7-6, as are the iodine appearance rates associated with an accident-initiated iodine spike.

6.7.2 Steamline Break Radiological Consequences

The complete severance of a main steamline outside containment is assumed to occur. The affected steam generator will rapidly depressurize and release iodine activity initially contained in the secondary coolant and primary coolant activity (iodines and noble gases) transferred via steam generator tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact steam generator and the activity transferred to the secondary coolant due to tube leakage is released to the atmosphere through either the atmospheric relief valves (ARVs) or the safety valves. The steamline break outside containment will bound any

break inside containment since the outside containment break provides a means for direct release to the environment. This section describes the assumptions and analyses performed to determine the offsite and Control Room doses resulting from the release of activity associated with this event.

6.7.2.1 Input Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 6.7-7.

The analysis of the MSLB radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analysis performed for the uprate follows this method as presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power. The activity available for release to the environment includes the iodine assumed to be initially present in the secondary coolant and the activity in the primary coolant (both iodine and noble gases) that could leak into the secondary coolant due to steam generator tube leakage.

Source Term

The iodine activity concentration of the secondary coolant at the time the MSLB occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of dose equivalent (DE) I-131.

The MSLB is analyzed for two iodine spiking cases: one in which there is a pre-existing iodine spike resulting in elevated primary coolant activity, and the other in which an iodine spike is assumed to be initiated by the accident. For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the MSLB, and has raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131. For the accident-initiated iodine spike case, the reactor trip associated with the MSLB creates an iodine spike in the RCS that increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 8 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted within 4 hours, and the accident-initiated spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the accident occurs is based on operation with a fuel defect level of 1.0 percent.

<u>Release Pathway</u>

The primary to secondary steam generator tube leakage rate is assumed to be at the Technical Specification limit of 150 gpd/SG. (This removes excess conservatism in the licensing submittal (Reference 1) analysis, which assumed a leak rate of 500 gpd/SG, to bound the Technical Specification limit of 150 gpd/SG. That analysis allowed for a potential relaxation of the Technical Specification limit without requiring re-analysis.)

The tube leakage in both the faulted and intact steam generators is assumed to persist for 72 hours, following initiation of the event.

The steam generator connected to the broken steamline is assumed to boil dry within 2 minutes following the MSLB. The entire liquid inventory of this steam generator is assumed to be steamed off and all of the iodine initially in this steam generator released to the environment. Also, iodine carried over to the faulted steam generator by tube leakage is assumed to be released directly to the environment, with no credit taken for iodine retention in the steam generator.

An iodine-partition factor in the intact steam generator of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) is used. Prior to reactor trip and concurrent loss-of-offsite power, an iodine-removal factor of 0.01 could be taken for steam released to the condenser, but this is conservatively ignored.

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

At 24 hours after onset of the accident, the Residual Heat Removal System (RHRS) is assumed to remove all decay heat, and there are no further steam releases to the atmosphere from the intact steam generator.

Within 72 hours after the accident, the Reactor Coolant System (RCS) has been cooled to below 212°F, and there are no further steam releases to the atmosphere from the faulted steam generator.

No fuel failure (departure from nucleate boiling [DNB] or melt) is calculated to occur for the MSLB event.

Control Room Isolation

In the event of an MSLB, the low steamline pressure safety injection (SI) setpoint will be reached shortly after event initiation. The SI signal causes the Control Room heating, ventilation, and air conditioning (HVAC) to switch from the normal-operation mode to the accident mode of operation. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 5 minutes after event initiation.

6.7.2.2 Acceptance Criteria

The offsite dose limit for an MSLB with a pre-accident iodine spike is 25 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is the guideline value of 10CFR50.67. For an MSLB with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is 10 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE per 10CFR50.67.

6.7.2.3 Results and Conclusions

The doses due to the MSLB with a pre-existing iodine spike are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Pre-accident Iodine Spike - SB	0.03	25
Pre-accident Iodine Spike - LPZ	0.01	25
Pre-accident Iodine Spike - Control Room	0.50	5

The doses due to the MSLB with an accident-initiated iodine spike are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Concurrent Iodine Spike - SB	0.06	2.5
Concurrent Iodine Spike - LPZ	0.02	2.5
Concurrent Iodine Spike - Control Room	1.00	5

The acceptance criteria are met.

The SB doses reported are for the worst 2-hour period, determined to be from 0 to 2 hours for the pre-accident iodine spike, and from 4 to 6 hours for the accident-initiated iodine spike.

6.7.3 Locked-Rotor Accident

An instantaneous seizure of a reactor coolant pump (RCP) rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop (RCL). Fuel-cladding damage may be predicted as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products transfer from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves, or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident, and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

6.7.3.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 6.7-8.

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analysis performed for the uprate follows this method as presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power.

<u>Source Term</u>

The analysis of the locked rotor radiological consequences assumes a pre-existing iodine spike in the RCS. For the pre-existing iodine spike, it is assumed that a reactor transient has occurred prior to the event that has raised the RCS iodine concentration to 60 μCi/gm of dose equivalent (DE) I-131.

The noble gas and alkali metal activity concentration in the primary coolant when the accident occurs is based on a fuel defect level of 1 percent. The iodine activity concentration of the secondary coolant when the locked rotor occurs is assumed to be 0.1 μCi/gm of dose equivalent (DE) I-131. The alkali metal activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be 10 percent of the primary side concentration.

As a result of the locked-rotor event, less than 50 percent of the fuel rods in the core undergo DNB. In the determination of the offsite and Control Room doses following the locked-rotor event presented in this report, it is conservatively assumed that 100 percent of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the primary coolant. Eight percent of the total core activity of iodine, 5 percent of the total core activity for noble gases, and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap and are released into the primary coolant.

Release Pathway

Activity is released to the environment by way of primary to secondary leakage and steaming from the secondary side to the environment. The primary to secondary steam generator tube leakage rate is assumed to be at the Technical Specification limit of 150 gpd/SG. This removes excess conservatism in the licensing submittal (Reference 1) analysis, which assumed a leak rate of 500 gpd/SG, to bound the Technical Specification limit for of 150 gpd/SG. That analysis allowed for a potential relaxation of the Technical Specification limit without requiring reanalysis.

The RHRS is assumed to remove all decay heat 8 hours into the accident, with no further releases to the environment after that time.

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) is used. This partition factor is applied to alkali metals. Prior to reactor trip and concurrent loss-of-offsite power, an iodine-removal factor of 0.01 could be taken for steam released to the condenser, but this is conservatively ignored.

All noble gas activity carried over to the secondary side through steam generation tube leakage is assumed to be immediately released to the outside atmosphere.

Control Room Isolation

It is assumed that the Control Room HVAC system begins in normal-operation mode. The activity level in the air supply duct gradually increases as activity builds up in the Control Room, and the concentration of activity in the steam generators (and consequently in the steam being released) increases. This causes a high-radiation signal to be generated. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 10 minutes after event initiation.

6.7.3.2 Acceptance Criteria

The offsite dose limit for a locked rotor is 2.5 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is 10 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE, per 10CFR50.67.

6.7.3.3 Results and Conclusions

The doses due to the locked rotor, including the 1.06 multiplier developed to bound variations in core average enrichment, core mass, and cycle length for this event, are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.50	2.5
LPZ	0.08	2.5
Control Room	1.40	5

The acceptance criteria are met.

The SB dose reported is for the worst 2-hour period, determined to be from 6 to 8 hours

6.7.4 Rod-Ejection Accident

It is assumed that a control rod drive mechanism (CRDM) pressure housing mechanical failure of has occurred, resulting in the ejection of a rod cluster control assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melting (pellet centerline) are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive primary coolant is assumed to leak from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the atmospheric relief valves, or the safety valves. Also, Iodine and alkali metal group activity is contained in the secondary coolant prior to the accident, and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident. Finally, radioactive primary coolant is discharged to the containment via spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

6.7.4.1 Input Parameters and Assumptions

Separate calculations are performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment, and the dose resulting from the leakage of activity to the secondary system and subsequent release to the environment. The total offsite and Control Room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered.

A summary of input parameters and assumptions is provided in Tables 6.7-9 and 6.7-10.

The analysis of the rod ejection radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analysis performed for the power uprate follows this method as presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power.

<u>Source Term</u>

Less than 15 percent of the fuel rods in the core undergo DNB as a result of the rod-ejection accident. In determining the offsite doses following a rod-ejection accident, it is conservatively assumed that 15 percent of the fuel rods in the core suffer sufficient damage such that all of their gap activity is released. Ten percent of the total core activity of iodine and noble gases, and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. In the calculation of activity releases from the failed/melted fuel, the maximum radial peaking factor of 1.7 was applied.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod-ejection accident. This amounts to 0.375 percent of the core, and the melting takes place in the centerline of the affected rods. The 0.375 percent of the fuel assumes that 15 percent of the rods in the core enter DNB. Of the rods that enter DNB, 50 percent are assumed to experience some melting of the fuel (7.5 percent of the core). Of the rods experiencing melting, 50 percent of the axial length of the rod is assumed to experience melting (3.75 percent of the core). It is further assumed that only 10 percent of the radial portion of the rod experiences melting (0.375 percent of the total core).

For both the containment leakage release path and the primary-to-secondary leakage release path, all noble gas and alkali metal activity released from the failed fuel (both gap activity and melted fuel activity) is available for release.

For the containment leakage release path, all of the iodine released from the gap of failed fuel and 25 percent of the activity released from melted fuel is available for release from containment.

For the primary-to-secondary leakage release path, all of the iodine released from the gap of failed fuel and 50 percent of the activity released from melted fuel is available for release from the RCS.

A pre-existing iodine spike in the reactor coolant is assumed to have increased the primary coolant iodine concentration to 60 µCi/gm of dose equivalent (DE) I-131 prior to the rod-ejection accident. The noble gas and alkali metal activity concentrations in the RCS when the accident occurs are based on operation with a fuel-defect level of 1 percent. The iodine activity concentration of the secondary coolant when the rod-ejection accident occurs is assumed to be equivalent to 0.1 µCi/gm of dose equivalent (DE) I-131. The alkali metal activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be 10 percent of the primary side concentration.

Iodine Chemical Form

Iodine in containment is assumed to be 4.85 percent elemental, 0.15 percent organic and 95 percent particulate. Iodine released from the secondary system is assumed to be 97 percent elemental, and 3 percent organic.

Release Pathways

When determining the offsite doses due to containment leakage, all of the RCS iodine, noble gas and alkali metal activity (from prior to the accident and resulting from the accident) is assumed to be in the containment.

The Containment System of the Kewaunee plant consists of three buildings: the reactor containment vessel, the shield building, and the auxiliary building. The containment and shield buildings are modeled as discrete volumes, which consider hold-up, removal, and decay. Since

no removal or hold-up is modeled for the auxiliary building, a separate volume representing the auxiliary building is not included in the model.

The containment is assumed to leak at the design-leak rate of 0.5 percent per day for the first 24 hours of the accident, and then to leak at half that rate (0.25 percent per day) for the remainder of the 30-day period following the accident considered in the analysis.

During the first 10 minutes of the accident, it is assumed that 90 percent of the activity leaking from the containment is discharged directly to the environment, and 10 percent enters the auxiliary building, where it is released through filters. After 10 minutes, only 1 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the auxiliary building, and 89 percent is assumed to pass into the shield building. The air discharged from the shield building is filtered to remove iodine and particulates. Additionally, once the shield building is brought to sub-atmospheric pressure at 30 minutes into the event, the iodine and particulates are subject to removal by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

When determining the offsite doses due to the primary-to-secondary steam generator tube leakage, all of the RCS iodine, noble gas, and alkali metal activity (from before the accident and resulting from the accident) is assumed to be in the primary coolant.

Primary-to-secondary tube leakage and steaming from the steam generators continues until the RCS pressure drops below the secondary pressure. A bounding time of 30 minutes was selected for this analysis, although the analysis shows that this would occur well before then.

The primary-to-secondary steam generator tube leakage rate is assumed to be at the Technical Specification limit of 150 gpd/SG. This removes excess conservatism in the licensing submittal (Reference 1) analysis, which assumed a leak rate of 500 gpd/SG, to bound the Technical Specification limit of 150 gpd/SG. That analysis allowed for a potential relaxation of the Technical Specification limit without requiring re-analysis. Although the primary-to-secondary pressure differential drops throughout the event, a constant leakage rate is assumed.

<u>Removal Coefficients</u>

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) is used. This partition factor is applied to alkali metals. Prior to reactor trip

and concurrent loss-of-offsite power, an iodine-removal factor of 0.01 could be taken for steam released to the condenser, but this is conservatively ignored. All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

For the containment leakage pathway, no credit is taken for sedimentation or plateout onto containment surfaces, or for containment spray operation that would remove airborne particulates and elemental iodine.

Control Room Isolation

The low pressurizer pressure SI setpoint will be reached within 60 seconds from event initiation. The SI signal causes the Control Room HVAC to switch from the normal-operation mode to the accident mode of operation. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 2.5 minutes after event initiation.

6.7.4.2 Acceptance Criteria

The offsite dose limit for a rod ejection is 6.3 rem TEDE, per Regulatory Guide 1.183 (Reference 2). This is ~25 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE, per 10CFR50.67.

6.7.4.3 Results and Conclusions

The offsite doses due to the rod-ejection accident, including the 1.06 multiplier developed to bound variations in core average enrichment, core mass, and cycle length for this event, are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.40	6.3
LPZ	0.09	6.3
Control Room	1.91	5

The acceptance criteria are met.

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

6.7.5 Steam Generator Tube Rupture Transient Offsite Dose Calculations

The calculation of the thermal-hydraulic results of the SGTR event is given in Section 6.3.

6.7.5.1 Input Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 6.7-11.

The analysis of the SGTR radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analysis performed for the uprate follows this method as presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power. The activity available for release to the environment includes the iodine assumed to be initially present in the secondary coolant and the activity in the primary coolant (both iodine and noble gases) that could leak into the secondary coolant due to steam generator tube leakage.

The SGTR is analyzed for two iodine spiking cases: one in which there is a pre-existing iodine spike resulting in elevated primary coolant activity, and the other in which an iodine spike is assumed to be initiated by the accident. For the pre-accident iodine-spike case, it is assumed that a reactor transient has occurred prior to the SGTR and has raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131. For the accident-initiated iodine-spike case, the reactor trip associated with the SGTR creates an iodine spike in the RCS, which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of dose equivalent (DE) I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 8 percent of the iodine in the fuel-cladding gap, the gap inventory would be depleted within 4 hours, and the accident-initiated spike is terminated at that time. (Regulatory Guide 1.183 (Reference 2) requires a spike multiplier of 500 for the steamline break, and allows a multiplier of 335 for SGTR. For this analysis, the SGTR spike factor was conservatively increased from 335 to 500 so that the same iodine appearance rates are used for the two events.)

The noble gas activity concentration in the RCS at the time the accident occurs is based on operation with a fuel-defect level of 1 percent. The iodine activity concentration of the

secondary coolant at the time the SGTR occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131.

Release Pathway

The break flow, flashed break flow, and steam release data presented in Table 6.3-2 of this document is used for the offsite and Control Room dose analysis.

The intact steam generator primary-to-secondary steam generator tube leakage rate is assumed to be at the Technical Specification limit of 150 gpd. This removes excess conservatism in the licensing submittal (Reference 1) analysis, which assumed a leak rate of 500 gpd, to bound the Technical Specification limit of 150 gpd/SG. That analysis allowed for a potential relaxation of the Technical Specification limit without requiring reanalysis.

An iodine-partition factor in the steam generators of 0.01 (curies iodine/gm steam)/(curies iodine/gm water) is used. Prior to reactor trip and concurrent loss-of-offsite power, an iodine-removal factor of 0.01 is taken for steam released to the condenser.

All noble gas activity carried over to the secondary side through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

Break-flow flashing fractions and steam-release rates from the intact and ruptured steam generator were calculated. The amount of break flow that flashes to steam is conservatively calculated, assuming that all break flow is from the hot-leg side of the break, and that the primary temperatures remain constant.

At 24 hours after the accident, the RHRS is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.

Control Room Isolation

The low-pressurizer pressure SI setpoint will be reached at ~2.9 minutes from event initiation. The SI signal causes the Control Room HVAC to switch from the normal-operation mode to the accident mode of operation. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 5 minutes after event initiation.

6.7.5.2 Acceptance Criteria

The offsite dose limit for a SGTR with a pre-accident iodine spike is 25 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is the guideline value of 10CFR50.67. For a SGTR with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is 10 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE per 10CFR50.67.

6.7.5.3 Results and Conclusions

The doses due to the SGTR with a pre-existing iodine spike are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Pre-Accident Iodine Spike - SB	1.30	25
Pre-Accident Iodine Spike - LPZ	0.30	25
Pre-Accident Iodine Spike - Control Room	3.10	5

The doses due to the SGTR with an accident-initiated iodine spike are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
Concurrent Iodine Spike - SB	0.80	2.5
Concurrent Iodine Spike - LPZ	0.20	2.5
Concurrent Iodine Spike - Control Room	1.00	5

The acceptance criteria are met.

The SB doses reported are for the worst 2-hour period, determined to be from 0 to 2 hours.

6.7.6 Large-Break Loss-of-Coolant Accident

An abrupt failure of a reactor coolant pipe is assumed to occur, and it is also assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (that is, melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident (DBA) that considers a single active failure. Activity from the core is released to the containment and then to the environment by containment leakage or leakage from the Emergency Core Cooling System (ECCS) as it recirculates sump solution outside the containment.

6.7.6.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Tables 6.7-12 and 6.7-13.

The analysis of the LBLOCA radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analysis performed for the power uprate follows this method as presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power.

Activity from the damaged core is released into the containment. The analysis considers the release of activity from the containment via containment leakage. In addition, once recirculation of the ECCS is established, activity in the sump solution may be released to the environment by means of leakage from ECCS equipment outside containment in the auxiliary building. The total offsite and Control Room doses are the sum of the doses resulting from each of the postulated release paths. The following sections address topics of significant interest.

6.7.6.1.1 Source Term

The reactor coolant activity is assumed to be insignificant compared with the release from the core, and is not included in the analysis.

Of the total core activity provided in Table 6.7-4, the following portions are released to the containment atmosphere and available for release to the environment via containment leakage:

- 100 percent of the noble gases (Xe, Kr) (5 percent in the gap and 95 percent in the fuel)

- 40 percent of the iodines (5 percent in the gap and 35 percent in the fuel)
- 30 percent of the alkali metals (Cs, Rb) (5 percent in the gap and 25 percent in the fuel)
- 5 percent of the tellurium metals (Te, Sb)
- 2 percent of the barium and strontium
- 0.25 percent of the noble metals (Ru, Rh, Mo, Tc)
- 0.05 percent of the cerium group (Ce, Pu, Np)
- 0.02 percent of the lanthanides (La, Zr, Nd, Nb, Pr, Y, Cm, Am)

The release of activity to containment occurs over a 1.8-hour interval. The gap activity is released in the first 30 minutes, and the fraction of the core activity that is released does so over the next 1.3 hours. A gap fraction of 5 percent is assumed for iodines, noble gases, and alkali metals. Gap activity of the other nuclides is not considered. With the exception of the iodines and noble gases, all activity released to containment is modeled as particulates. The iodine in containment is modeled as 4.85 percent elemental, 0.15 percent organic, and 95 percent particulate. For ECCS leakage considerations, the iodine activity that becomes airborne after being released by the leakage is modeled as 97 percent elemental and 3 percent organic.

For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay, or leakage from the containment. For the ECCS leakage analysis, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS.

6.7.6.1.2 Containment Modeling

The Containment System of the KNPP consists of three buildings: the reactor containment vessel, the shield building, and the auxiliary building. The containment and shield buildings are modeled as discrete volumes that consider hold-up, removal, and decay. Since no hold-up is modeled for the auxiliary building, a separate volume representing the auxiliary building is not included in the model.

The containment is assumed to leak at the design-leak rate of 0.5 percent per day for the first 24 hours of the accident, and then to leak at half that rate (0.25 percent per day) for the remainder of the 30-day period following the accident considered in the analysis.

During the first 10 minutes of the accident, it is assumed that 90 percent of the activity leaking from the containment is discharged directly to the environment and 10 percent enters the auxiliary building where it is released through filters. After 10 minutes, only 1.0 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the auxiliary building, and 89 percent is assumed to pass into the shield building. The air discharged from the shield building is filtered to remove iodine and particulates. Additionally, once the shield building is brought to sub-atmospheric pressure at 30 minutes into the event, the iodine and particulates are subject to removal by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

6.7.6.1.3 Activity Removal from the Containment Atmosphere

The removal of elemental iodine from the containment atmosphere is accomplished only by containment sprays and radioactive decay. The removal of particulates from the containment atmosphere is accomplished by containment sprays, sedimentation, and radioactive decay. The noble gases and the organic iodine are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate following the LOCA. Injection spray is credited with no startup delay. Earlier spray actuation is conservative since it results in earlier spray termination. There is no benefit from the earlier actuation since there is little activity in the containment at the time the sprays start. When the refueling water storage tank (RWST) drains to a predetermined setpoint level, the operators switch to recirculation of sump liquid to provide a source for the sprays. Switchover to recirculation spray is not credited in the analysis and all spray is assumed to be terminated when the RWST drains down. The analysis conservatively assumed that the sprays are terminated 0.91 hours from the start of the event.

Containment Spray Removal of Elemental Iodine

The *Standard Review Plan* (Reference 7) identifies a methodology to determine spray removal of elemental iodine. The removal rate constant is determined by:

$$\lambda_s = 6K_gTF/VD$$

where

λ_s = Elemental iodine removal rate constant due to spray removal, hr^{-1}

K_g = Gas phase mass transfer coefficient, ft/min

T = Time of fall of the spray drops, min

F = Volume flow rate of sprays, ft^3/hr

V = Containment sprayed volume, ft^3

D = Mass-mean diameter of the spray drops, ft

The upper limit specified for this model is 20 hr^{-1}.

Parameters are listed below and were chosen to bound the current plant configuration:

K_g = 9.84 ft/min

T = 13 sec

F = 1148 gpm

V = 1.32E6 ft^3

D = 0.121 cm

These parameters and appropriate conversion factors were used to calculate the elemental spray removal coefficients. The calculated value is above the identified upper limit of 20 hr^{-1}.

Removal of elemental iodine from the containment atmosphere is assumed to be terminated when the airborne inventory drops to 0.5 percent of the total elemental iodine released to the containment (this is a DF of 200). With the Regulatory Guide 1.183 (Reference 2) source term methodology this is interpreted as being 0.5 percent of the total inventory of elemental iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis, this does not occur before spray termination at 0.91 hours.

Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Standard Review Plan (Reference 7).

The first order spray removal rate constant for particulates is written as follows:

$\lambda_p = 3hFE/2VD$

where

λ_p = Particulate removal rate constant due to spray removal, hr^{-1}

h = Drop Fall Height, ft

F = Spray Flow Rate, ft^3/hr

V = Volume Sprayed, ft^3

E = Single Drop Collection Efficiency

D = Average Spray Drop Diameter, ft

Parameters are listed below and were chosen to bound the current plant configuration:

H = 150 ft

F = 1148 gpm

V = 1.32E6 ft^3

The E/D term depends upon the particle size distribution and spray drop size. It is conservative to use 10 m^{-1} for E/D until the point is reached when the inventory in the atmosphere is reduced to 2 percent of its original (DF of 50). With the Regulatory Guide 1.183 (Reference 2) source term methodology this is interpreted as being 2 percent of the total inventory particulate iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases.

These parameters and the appropriate conversion factors were used to calculate the particulate spray removal coefficients. A conservative value of 4.5 hr^{-1} was used in the analysis. The airborne inventory does not drop to 2 percent of the total particulate iodine released to the containment (this is a DF of 50) before spray termination at 0.91 hours.

The discussion of spray removal coefficients above includes a reduction in assumed spray flow rate relative to that modeled in Reference 1 (that is, 1148 gpm rather than 1300 gpm), to bound potential pump degradation.

6.7.6.1.4 Sedimentation Removal of Particulates

During spray operation, no credit is taken for sedimentation removal of particulates, although it would take place. It is assumed that containment spray operation is terminated at 0.91 hours. Recirculation sprays are not credited. Credit is taken for sedimentation removal of particulates after spray termination. As discussed in the licensing submittal for implementing alternative

source term for the KNPP (Reference 1), the analysis credits a sedimentation coefficient of 0.1 h^{-1}.

6.7.6.1.5 Emergency Core Cooling System Leakage

When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. There are two pathways considered for the ECCS recirculation leakage. One is the leakage directly into the auxiliary building and the other is back-leakage into the RWST. Although recirculation is not initiated until the RWST has drained to the pre-determined setpoint level (at about 0.91 hours) the analysis conservatively considers leakage from the start of the event.

The leakage to the auxiliary building is 12 gallon/hr. (This is a more conservative value than assumed in the licensing submittal [Reference 1] analysis.) As discussed in the licensing submittal, this analysis models a conservative airborne fraction of 10 percent when the sump temperature is above 212°F. Once the sump solution temperature drops below 212°F, the airborne fraction is reduced to 1 percent. The reduction in airborne fraction is conservatively delayed until 3 hours from the start of the event. This delay is longer than that modeled in the licensing submittal (Reference 1) as it is selected to bound the results of containment response analyses performed for the power Uprate Program (Section 6.4).

RHR back-leakage to the RWST is assumed at a rate of 3 gpm for the first 24 hours, and 1.5 gpm for the remainder of the event. It is assumed that 1 percent of the iodine contained in the leak flow becomes airborne. The 1-percent value is applied even when the sump is above 212°F since any incoming water would be cooled by the water remaining in the RWST. The RWST vents to the auxiliary building.

It is assumed that half the iodine activity that becomes airborne in the auxiliary building from the two leak sources is removed by plateout on surfaces. Releases from the auxiliary building are subject to filtration by the auxiliary building special ventilation system.

6.7.6.1.6 Control Room Isolation

In the event of a LBLOCA, the low pressurizer pressure SI setpoint will be reached shortly after event initiation. The SI signal causes the Control Room HVAC to switch from the normal-operation mode to the accident mode of operation. It is conservatively assumed that the Control

Room HVAC does not fully enter the accident mode of operation until 2 minutes after event initiation.

6.7.6.2 Acceptance Criteria

The offsite dose limit for a LOCA is 25 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE per 10CFR50.67.

6.7.6.3 Results and Conclusions

The calculated offsite and Control Room doses due to the three release paths considered for the LBLOCA, including the 1.03 multiplier developed to bound variations in core average enrichment, core mass, and cycle length for this event, are:

	SB Dose (rem TEDE)	LPZ Dose (rem TEDE)	Control Room Dose (rem TEDE)
Containment Leakage	1.242	0.196	3.901
ECCS Leakage	1.864E-02	3.621E-03	7.946E-02
RWST Back-Leakage	4.316E-02	1.218E-02	5.616E-01
Total	1.304	0.212	4.542

Subsection 7.6.9 discusses the calculation of the direct and skyshine Control Room dose. The calculated dose for the 30-day duration considered in this analysis is 0.036 rem. This dose is added to the Control Room TEDE dose discussed above.

The total offsite and Control Room doses due to the LBLOCA are reported as:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	1.31	25
LPZ	0.22	25
Control Room	4.58	5

The acceptance criteria are met.

The SB dose reported is for the worst 2-hour period, determined to be from 1.8 to 3.8 hours.

6.7.7 Gas Decay Tank Rupture Radiological Consequences

For the GDT rupture analysis, there is assumed to be a failure that results in the release of the contents of one GDT.

6.7.7.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Tables 6.7-14.

The analysis performed for the uprate follows the method presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power.

The inventory of gases in the tank are based on operation of the plant with 1 percent fuel defects and with no purge of activity from the VCT to the GDT during the cycle. This is followed by shutdown of the plant and degassing of the primary coolant to the tank. The gas decay tank inventory resulting from this unit shutdown degassing process is provided in Table 6.7-14.

A failure in the gaseous waste processing system is assumed to result in release of the tank inventory with a release duration of 5 minutes.

<u>Control Room Isolation</u>

It is assumed that the Control Room HVAC system begins in normal-operation mode. The activity level in the air supply duct causes a high radiation signal almost immediately. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 30 seconds after event initiation.

Although the maximum unfiltered leakage rate of 200 scfm listed in Table 6.7-3 is limiting for all other events, a lower rate was determined to be conservative for the gas decay tank rupture and VCT rupture. After the 5-minute activity release period the air outside the Control Room has no activity, while the Control Room has a high concentration of noble gases. Since there is no forced makeup flow the only flow that dilutes the concentration of activity in the Control Room is

the inleakage. Minimizing this inleakage maximizes the dose. The analysis conservatively neglected all inleakage in the calculation of the Control Room doses. This is a conservative change relative to the modeling used in the licensing submittal (Reference 1) analysis, which assumed a 200-scfm inleakage for Control Room doses.

6.7.7.2 Acceptance Criteria

The offsite dose limits for a GDT rupture are assumed to be the same as those for the FHA, that is, 6.3 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is ~25 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE per 10CFR50.67.

6.7.7.3 Results and Conclusions

The doses due to the GDT rupture are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.10	6.3
LPZ	0.02	6.3
Control Room	0.80	5

The acceptance criteria are met.

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

6.7.8 Volume Control Tank Rupture

For the VCT rupture, a failure is assumed that results in the release of the contents of the tank plus the noble gases and a fraction of the iodines from the letdown flow until the letdown path is isolated.

6.7.8.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.7-15.

The analysis performed for the uprate follows the method presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power.

The inventory of gases in the tank is based on continuous operation with 1.0 percent fuel defects and without any purge of the gas space. The inventory of iodine in the tank is based on operation of the plant with 1.0 percent fuel defects and with 90 percent of the iodine removed by the letdown demineralizer.

As a result of the accident, all of the noble gas in the tank and 1.0 percent of the iodine in the tank liquid are assumed to be released to the atmosphere over a period of 5 minutes.

After event initiation, letdown flow to the VCT continues at the maximum flow rate of 88 gpm (maximum letdown flow plus 10-percent uncertainty) for 5 minutes when the letdown line is assumed to be isolated. The primary coolant noble gas activities used in the VCT rupture dose calculations are based on operation with 1 percent fuel defects. The primary coolant iodine activity is conservatively assumed to be at the pre-existing iodine spike level of 60 μCi/gram dose equivalent (DE) I-131, which is reduced by 90 percent by the letdown demineralizer. All of the noble gas and 1.0 percent of the iodine in the letdown flow are assumed to be released to the environment.

Control Room Isolation

It is assumed that the Control Room HVAC system begins in normal-operation mode. The activity level in the air supply duct causes a high-radiation signal almost immediately. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 30 seconds after event initiation.

Although the maximum unfiltered leakage rate of 200 scfm listed in Table 6.7-3 is limiting for all other events, a lower rate was determined to be conservative for the gas decay tank rupture and VCT rupture. After the 5-minute activity release period, the air outside the Control Room has no activity, while the Control Room has a high concentration of noble gases. Since there is no forced makeup flow, the only flow that dilutes the concentration of activity in the Control Room is the inleakage. Minimizing this inleakage maximizes the dose. The analysis conservatively neglected all inleakage in the calculation of the Control Room doses. This is a conservative

change relative to the modeling used in the licensing submittal (Reference 1) analysis, which assumed a 200-scfm inleakage for Control Room doses.

6.7.8.2 Acceptance Criteria

The offsite dose limits for a VCT rupture are assumed to be the same as those for the FHA, that is, 6.3 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is ~25 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE per 10CFR50.67.

6.7.8.3 Results and Conclusions

The doses due to the VCT rupture are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.10	6.3
LPZ	0.01	6.3
Control Room	0.40	5

The acceptance criteria are met.

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

6.7.9 Fuel-Handling Accident

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the auxiliary building. Activity released from the damaged assembly is released to the outside atmosphere through either the Containment Purge System or the Spent Fuel Pool Ventilation System.

6.7.9.1 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 6.7-16.

The analysis of the FHA radiological consequences uses the analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 2). The analysis performed for the uprate follows this method as presented in the licensing submittal for implementing alternative source term for the KNPP (Reference 1), with changes made to reflect the increased power.

All activity released from the fuel pool is assumed to be released to the atmosphere in 2 hours, using an exponential release model with higher releases in the initial periods since this is conservative for the Control Room doses. No credit is taken for operation of the Spent Fuel Pool Ventilation System in the auxiliary building. No credit is taken for isolation of containment for the FHA in containment. Since the assumptions and parameters for a FHA inside containment are identical to those for a FHA in the auxiliary building, the radiological consequences are the same regardless of the location of the accident.

Source Term

The calculation of the radiological consequences following a FHA uses gap fractions of 8 percent for I-131, 10 percent for Kr-85, and 5 percent for all other nuclides.

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.7 times the core average power. The activity calculated for the third transition cycle is conservatively increased by 6 percent to bound variations in core average enrichment, core mass, and cycle length (Table 6.7-17).

The decay time used in the analysis is 100 hours. Thus, the analysis supports the Technical Specifications limit of 100 hours decay time prior to fuel movement.

Iodine Chemical Form

Iodine species in the pool is 99.85 percent elemental and 0.15 percent organic. This is based on the split leaving the fuel of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine and 0.15 percent organic iodine. It assumed that all CsI is dissociated in the water and the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. Thus, 99.85 percent of the iodine released is elemental.

Pool Scrubbing Removal of Activity

Per the Technical Specifications, it is assumed that there is a minimum of 23 feet of water above the fuel. With this water depth, the overall pool decontamination factor (DF) for iodine is 200. The DF for organic iodine and noble gases is 1.0. (The elemental iodine-scrubbing factor is conservatively set to provide the overall effective DF of 200 specified in Regulatory Guide 1.183 [Reference 2]. By back calculating, it is approximately 286, which is well below the value of 500 indicated in the Regulatory Guide, and conservatively accounts for increased rod internal pressures above 1200 psig.)

The cesium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

The split between elemental and organic iodine leaving the pool has no impact on the analysis since the Control Room filter efficiencies are the same, and no other filtration is credited.

Filtration of Release Paths

No credit is taken for removing iodine by filters, nor is credit taken for isolating release paths.

Although the containment purge will be automatically isolated on a purge-line high-radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2-hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

Control Room Isolation

It is assumed that the Control Room HVAC system begins in normal-operation mode. The activity level in the air supply duct causes a high- radiation signal almost immediately. It is conservatively assumed that the Control Room HVAC does not fully enter the accident mode of operation until 1 minute after event initiation.

6.7.9.2 Acceptance Criteria

The offsite-dose limit for a FHA is 6.3 rem TEDE per Regulatory Guide 1.183 (Reference 2). This is ~25 percent of the guideline value of 10CFR50.67. The limit for the Control Room dose is 5.0 rem TEDE per 10CFR50.67.

6.7.9.3 Results and Conclusions

The doses due to the FHA are:

Case	TEDE Dose (rem)	Acceptance Criteria (rem TEDE)
SB	0.70	6.3
LPZ	0.11	6.3
Control Room	1.00	5

The acceptance criteria are met.

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

6.7.10 References

1. Letter from M. E. Warner (NMC) to Document Control Desk (NRC), *Revision to the Design Basis Radiological Analysis Accident Source Term*, March 19, 2002.

   - Letter from T. Couto (NMC) to Document Control Desk (NRC), Response to Request for Additional Information Related to Proposed Revision to the Kewaunee Nuclear Power Plant Design-Basis Radiological Analysis Accident Source Term, September 13, 2002.

   - Letter from T. Couto (NMC) to Document Control Desk (NRC), Kewaunee Nuclear Power Plant Alternate Source Term Report – Rev. 2, September 13, 2002.

2. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000.

3. EPA Federal Guidance Report No. 11, EPA-520/1-88-020, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, September 1988.

4. International Commission on Radiological Protection, *Radionuclide Transformations, Energy and Intensity of Emissions*, ICRP Publication 38, 1983.

5. International Commission on Radiological Protection, *Limits for Intakes of Radionuclides by Workers*, ICRP Publication 30, 1979.

6. K. G. Murphy and K. M. Campe, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,* Proceedings of the Thirteenth AEC Air Cleaning Conference held August 1974.

7. NUREG-0800, Standard Review Plan 6.5.2, *Containment Spray as a Fission Product Cleanup System*, Rev. 2, December 1988.

Table 6.7-1 Nuclide Parameters				
Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	Average Gamma Disintegration Energies (Mev/dis)	Noble Gas DCF (rem· m^3 /Ci · sec)
I-131	0.00359	3.29E4	0.38	NA
I-132	0.303	3.81E2	2.2	NA
I-133	0.0333	5.85E3	0.6	NA
I-134	0.791	1.31E2	2.6	NA
I-135	0.105	1.23E3	1.4	NA
Kr-85m	0.155	NA	NA	0.0307
Kr-85	7.37E-6	NA	NA	0.000484
Kr-87	0.547	NA	NA	0.146
Kr-88	0.248	NA	NA	0.37
Xe-131m	0.00241	NA	NA	0.00152
Xe-133m	0.0130	NA	NA	0.00553
Xe-133	0.00546	NA	NA	0.00624
Xe-135m	2.72	NA	NA	0.0775
Xe-135	0.0756	NA	NA	0.0482
Xe-138	2.93	NA	NA	0.198
Cs-134	3.84E-5	4.62E4	1.55	NA
Cs-136	2.2E-3	7.33E3	2.16	NA
Cs-137	2.64E-6	3.19E4	0.564	NA
Rb-86	1.55E-3	6.63E3	0.0945	NA

Notes:

CEDE = Committed effective dose equivalent

DCF = Dose conversion factor

Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	Average Gamma Disintegration Energies (Mev/dis)	Noble Gas DCF (rem· m^3 /Ci · sec)
Te-127m	2.65E-4	2.15E4	0.0112	NA
Te-127	7.41E-2	3.18E2	4.86E-3	NA
Te-129m	8.6E-4	2.39E4	0.0375	NA
Te-129	0.598	9.0E1	0.0591	NA
Te-131m	2.31E-2	6.4E3	1.42	NA
Te-132	8.86E-3	9.44E3	0.233	NA
Sb-127	7.5E-3	6.04E3	0.688	NA
Sb-129	0.16	6.44E2	1.44	NA
Sr-89	5.72E-4	4.14E4	8.45E-5	NA
Sr-90	2.72E-6	1.3E6	0.0	NA
Sr-91	0.073	1.66E3	0.693	NA
Sr-92	0.256	8.1E2	1.34	NA
Ba-139	0.502	1.7E2	0.043	NA
Ba-140	2.27E-3	3.74E3	0.182	NA
Ru-103	7.35E-4	8.95E3	0.468	NA
Ru-105	0.156	4.55E2	0.775	NA
Ru-106	7.84E-5	4.77E5	0.0	NA
Rh-105	1.96E-2	9.56E2	0.078	NA
Mo-99	1.05E-2	3.96E3	0.15	NA
Tc-99m	0.115	3.3E1	0.126	NA

Table 6.7-1 (Cont.)

Nuclide Parameters

Notes:

CEDE = Committed effective dose equivalent

DCF = Dose conversion factor

Table 6.7-1 (Cont.) Nuclide Parameters				
Nuclide	**Decay Constant (hr^{-1})**	**CEDE DCF (rem/Ci inhaled)**	**Average Gamma Disintegration Energies (Mev/dis)**	**Noble Gas DCF (rem· m^3 /Ci · sec)**
Ce-141	8.89E-4	8.96E3	0.076	NA
Ce-143	0.021	3.39E3	0.282	NA
Ce-144	1.02E-4	3.74E5	0.021	NA
Pu-238	9.02E-7	3.92E8	1.81E-3	NA
Pu-239	3.29E-9	4.3E8	8.08E-4	NA
Pu-240	1.21E-8	4.3E8	1.73E-3	NA
Pu-241	5.5E-6	8.26E6	2.54E-6	NA
Np-239	0.0123	2.51E3	0.172	NA
Y-90	1.08E-2	8.44E3	1.7E-6	NA
Y-91	4.94E-4	4.89E4	3.61E-3	NA
Y-92	0.196	7.80E2	0.251	NA
Y-93	0.0686	2.15E3	0.0889	NA
Nb-95	8.22E-4	5.81E3	0.766	NA
Zr-95	4.51E-4	2.37E4	0.739	NA
Zr-97	4.1E-2	4.33E3	0.179	NA
La-140	1.72E-2	4.85E3	2.31	NA
La-141	0.176	5.81E2	0.0427	NA
La-142	0.45	2.53E2	2.68	NA
Nd-147	2.63E-3	6.85E3	0.14	NA
Pr-143	2.13E-3	1.09E4	8.9E-9	NA
Am-241	1.83E-7	4.44E8	0.0324	NA
Cm-242	1.77E-4	1.73E7	1.83E-3	NA
Cm-244	4.37E-6	2.48E8	1.7E-3	NA

Notes:

CEDE = Committed effective dose equivalent

DCF = Dose conversion factor

Table 6.7-2 Offsite Breathing Rates and Atmospheric Dispersion Factors	
Time	**Offsite Breathing Rates (m^3/sec)**
0 - 8 hours	3.47E-4
8 - 24 hours	1.75E-4
>24 hours	2.32E-4
	Offsite Atmospheric Dispersion Factors (sec/m^3)
SB[1]	2.232E-4
LPZ	
0 - 2 hours	3.977E-5
2 - 24 hours	4.100E-6
1 - 2 days	2.427E-6
> 2 days	4.473E-7

Note:

1. This SB atmospheric dispersion factor is conservatively applied during all time intervals in the determination of the limiting 2-hour period

Table 6.7-3 Control Room Parameters	
Breathing Rate - Duration of the Event	3.47E-4 m^3/sec
Control Room Volume	127,600 ft^3
Atmospheric Dispersion Factors 0 - 8 hours 8 - 24 hours 1 - 4 days 4 - 30 days	 2.93E-3 m^3/sec 1.73E-3 m^3/sec 6.74E-4 m^3/sec 1.93E-4 m^3/sec
Occupancy Factors[1] 0 - 24 hours 1 - 4 days 4 - 30 days	 1.0 0.6 0.4
Normal Ventilation Flow Rates Filtered Makeup Flow Rate Filtered Recirculation Flow Rate Unfiltered Makeup Flow Rate Unfiltered Recirculation Flow Rate	 0.0 scfm 0.0 scfm 2500 scfm (±10%) (Not modeled - no impact on analyses)
Emergency Ventilation System Flow Rates Filtered Makeup Air Flow Rate Filtered Recirculation Flow Rate Unfiltered In-leakage	 0.0 scfm 2500 scfm (±10%) 200 scfm
Filter Efficiencies Elemental Organic Particulate	 90% 90% 99%
R-23 Sensitivity Setpoint for Control Room Isolation Location of R-23	2.32E7 (cpm/µCi/cc for Xe-133) 1.0E4 (cpm) R-23 is located at a junction of the intake and the recirculation ducts such that it monitors the mixed air stream
Delay to Initiate Switchover of HVAC from Normal Operation to Emergency Operation after SI Signal	63 seconds
Delay for Switchover of HVAC from Normal Operation to Emergency Operation after Receipt of an Isolation Signal	<10 seconds

Note:

1. *These occupancy factors (from Reference 6) have been conservatively incorporated in the atmospheric dispersion factors. This is conservative, since it does not allow the benefit of reduced occupancy for activity already present in the Control Room from earlier periods.*

Table 6.7-4

Core Total Fission Product Activities
Based on 1782.6 MWt (100.6% of 1772 MWt)

Isotope	Activity (Ci)
I-131	4.76E7
I-132	6.91E7
I-133	9.83E7
I-134	1.08E8
I-135	9.18E7
Kr-85m	1.31E7
Kr-85	5.39E5
Kr-87	2.53E7
Kr-88	3.56E7
Xe-131m	5.32E5
Xe-133m	2.88E6
Xe-133	9.42E7
Xe-135m	1.91E7
Xe-135	2.61E7
Xe-138	8.16E7
Cs-134	9.26E6
Cs-136	2.64E6
Cs-137	5.75E6
Rb-86	1.05E5
Te-127m	6.51E5
Te-127	5.01E6
Te-129m	2.22E6
Te-129	1.50E7
Te-131m	6.90E6
Te-132	6.80E7
Sb-127	5.05E6
Sb-129	1.53E7

Table 6.7-4 (Cont.)

Core Total Fission Product Activities
Based on 1782.6 MWt (100.6% of 1772 MWt)

Isotope	Activity (Ci)
Sr-89	4.82E7
Sr-90	4.26E6
Sr-91	5.97E7
Sr-92	6.44E7
Ba-139	8.81E7
Ba-140	8.48E7
Ru-103	7.16E7
Ru-105	4.81E7
Ru-106	2.38E7
Rh-105	4.45E7
Mo-99	9.08E7
Tc-99m	7.96E7
Ce-141	8.06E7
Ce-143	7.52E7
Ce-144	6.17E7
Pu-238	1.79E5
Pu-239	1.82E4
Pu-240	2.52E4
Pu-241	5.89E6
Np-239	9.50E8

Table 6.7-4 (Cont.)

Core Total Fission Product Activities
Based on 1782.6 MWt (100.6% of 1772 MWt)

Isotope	Activity (Ci)
Y-90	4.42E6
Y-91	6.18E7
Y-92	6.47E7
Y-93	7.42E7
Nb-95	8.27E7
Zr-95	8.22E7
Zr-97	8.13E7
La-140	9.21E7
La-141	8.05E7
La-142	7.80E7
Nd-147	3.21E7
Pr-143	7.27E7
Am-241	7.13E3
Cm-242	1.53E6
Cm-244	1.57E5

Table 6.7-5

RCS Coolant Concentrations
Based on 1.0 μCi/gm DE I-131 for Iodines[1]
and 1-% Fuel Defects for Noble Gases and Alkali Metals

Nuclide	Activity (μCi/gm)
I-131	0.780
I-132	0.793
I-133	1.164
I-134	0.161
I-135	0.637
Kr-85m	1.73
Kr-85	8.60
Kr-87	1.13
Kr-88	3.28
Xe-131m	3.04
Xe-133m	3.44
Xe-133	242.0
Xe-135m	0.501
Xe-135	8.69
Xe-138	0.628
Cs-134	2.86
Cs-136	3.22
Cs-137	2.16
Rb-86	0.0316

Note:

1. Iodine concentrations are converted to DE I-131 using the DCFs in ICRP-30 (Reference 5) for direct thyroid doses.

Table 6.7-6 Iodine Spiking Data		
Isotope	**Primary Coolant Concentration for Pre-Existing Spike (μCi/gm)**	**Iodine Appearance Rate into Primary Coolant for Accident-Initiated Spike (Ci/min)**
I-131	46.8	151
I-132	47.6	394
I-133	69.8	260
I-134	9.7	160
I-135	38.2	189

Table 6.7-7

Assumptions Used for Steamline Break Dose Analysis

Nuclide Parameters	See Table 6.7-1
Primary Coolant Noble Gas Activity prior to Accident	1.0% Fuel Defect Level (See Table 6.7-5)
Primary Coolant Iodine Activity prior to Accident	
Pre-Existing Spike	60 μCi/gm of DE I-131 (See Table 6.7-6)
Accident-Initiated Spike	1.0 μCi/gm of DE I-131 (see Table 6.7-5)
Primary Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate (See Table 6.7-6)
Duration of Accident-Initiated Spike	4.0 hours
Secondary Coolant Iodine Activity prior to Accident	0.1 μCi/gm of DE I-131 (1/10 of Table 6.7-5 values)
Faulted SG Tube Leak Rate during Accident	150 gpd
Intact SG Tube Leak Rate during Accident	150 gpd
SG Iodine Partition Factor	
Intact SG	0.01
Faulted SG	1.0
Time to Cool RCS Below 212°F and Stop Releases from Faulted SG	72 hours
Steam Release from Intact SG to Environment	
0-2 hours	222,000 lbm
2-8 hours	424,000 lbm
8-24 hours	614,000 lbm
Steam Release from Faulted SG to Environment (during first 2 minutes)	161,000 lbm
Primary Coolant Mass	1.19E8 gm
Intact Steam Generator Secondary Mass	84,000 lbm
Faulted Steam Generator Secondary Mass	161,000 lbm
Offsite Breathing Rates	See Table 6.7-2
Offsite Atmospheric Dispersion Factors	See Table 6.7-2
Control Room Model	See Table 6.7-3
Time to start Crediting Emergency Control Room HVAC	5 minutes

Table 6.7-8

Locked Rotor Accident Input Parameters and Assumptions

Source Term	
Core Activity	See Table 6.7-4
Fission Product Gap Fractions	
I-131	8%
Kr-85	10%
Other Iodines and Noble Gases	5%
Alkali Metals	12%
Nuclide Parameters	See Table 6.7-1
Fraction of Fuel Rods in Core Failing	100%
Iodine Chemical Form	97% elemental, 3% organic
Primary Coolant Activity before Fuel Failure	
Iodines	60 μCi/gm DE I-131 (see Table 6.7-6)
Noble Gases and Alkali Metals	Based on operation with 1.0% fuel defects (see Table 6.7-5)
Secondary Coolant Iodine Activity at Beginning of Event	0.1 μCi/gm DE I-131 (10% of Table 6.7-5 values)
Secondary Alkali Metal Activity at Beginning of Event	10% of Table 6.7-5 values
Release Path	
Primary Coolant Mass	1.19E8 gm
Secondary Coolant Mass	7.89E7 gm
Primary-to-Secondary Leak Rate	150 gal/day/SG
Steaming Released from the Secondary Side	
0 - 2 hr	210,000 lbm
2 - 8 hr	455,000 lbm
Steaming Partition Coefficient	0.01
Termination of Releases	8 hours
Offsite Atmospheric Dispersion Factors (sec/m^3)	See Table 6.7-2
Offsite Breathing Rates	See Table 6.7-2
Control Room Model	See Table 6.7-3
Time to Start Crediting Emergency Control Room HVAC	10 minutes

Table 6.7-9

Assumptions Used for Rod-Ejection Accident

<u>Source Term</u>	
Core Activity	See Table 6.7-4
Nuclide Parameters	See Table 6.7-1
Radial Peaking Factor	1.7
Fission Product Gap Fractions	
Iodines and Noble Gases	10%
Alkali Metals	12%
Fraction of Activity Released from Melted Fuel	
Noble Gases and Alkali Metals	100%
Iodines	25% for containment leakage release path 50% for steam generator steaming release path
RCS Iodines	60 μCi/gm DE I-131 (see Table 6.7-6)
RCS Noble Gas and Alkali Metals	1% fuel defects (see Table 6.7-5)
Secondary Coolant Iodine Activity	0.1 μCi/gm DE I-131 (10% of values in Table 6.7-5)
Secondary Alkali Metal Activity	10% of values in Table 6.7-5
<u>Containment Leakage Release Path</u>	
Iodine Chemical Form	4.85% elemental, 0.15% organic, and 95% particulate
Removal Coefficients	None assumed
Containment, Shield Building, and Auxiliary Building Modeling	See Table 6.7-10
<u>Steam Generator Steaming Release Path</u>	
Primary Coolant Mass	1.19E8 gm
Secondary Coolant Mass	7.89E7 gm
Primary-to-Secondary Leak Rate	150 gal/day/SG
Steaming Rate from the Secondary Side	
0 – 200 seconds 200 – 1800 seconds > 1800 seconds	800 lb/sec 100 lb/sec 0 lb/sec
Iodine Chemical Form	97% elemental, 3% organic
Steaming Partition Coefficient	0.01
<u>Offsite Atmospheric Dispersion Factors (sec/m^3)</u>	See Table 6.7-2
<u>Offsite Breathing Rates</u>	See Table 6.7-2
<u>Control Room Model</u>	See Table 6.7-3
Time to Start Crediting Emergency Control Room HVAC	2.5 minutes

Table 6.7-10

Containment, Shield Building, and Auxiliary Building Modeling Used for Rod-Ejection Accident

Containment Net-Free Volume	1.32E6 ft^3
Shield Building Volume	3.74E5 ft^3
Shield Building Participation Fraction	0.5
Auxiliary Building Volume	Not modeled, no holdup credited
Containment Leak Rates	
0 – 24 hours	0.5 (weight %/day)
> 24 hours	0.25 (weight %/day)
Containment Leak Path Fractions	
0 – 10 minutes	
Through Shield Building	0.0
Through Auxiliary Building SV	0.1
Direct to Environment	0.9
> 10 minutes	
Through Shield Building	0.89
Through Auxiliary Building SV	0.1
Direct to Environment	0.01
Shield Building Air Flows	
0 – 10 minutes	
Shield Building to Environment	Not applicable
Shield Building Recirculation	Not applicable
10 minutes – 30 minutes	
Shield Building to Environment	6000±10% scfm
Shield Building Recirculation	0.0 scfm
> 30 minutes	
Shield Building to Environment	3100 scfm
Shield Building Recirculation	2300 scfm
Spray and Sedimentation Removal in Containment	Not credited
Shield Building and Auxiliary Building Filter Efficiencies	
Elemental	90%
Organic	90%
Particulate	99%

Table 6.7-11

Assumptions Used for SGTR Dose Analysis

Nuclide Parameters	See Table 6.7-1
Primary Coolant Noble Gas Activity prior to Accident	1.0% fuel defect level (See Table 6.7-5)
Primary Coolant Iodine Activity prior to Accident	
Pre-Existing Spike	60 μCi/gm of DE I-131 (See Table 6.7-6)
Accident-Initiated Spike	1.0 μCi/gm of DE I-131 (see Table 6.7-5)
Primary Coolant Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rate (See Table 6.7-6)
Duration of Accident-Initiated Spike	4.0 hours
Secondary Coolant Iodine Activity prior to Accident	0.1 μCi/gm of DE I-131 (1/10 of Table 6.7-5 values)
Ruptured SG Steam Releases	See Table 6.3-2
Ruptured SG Break Flow Rate	See Table 6.3-2
Break-Flow Flashing Fractions	See Table 6.3-2
Intact SG Tube Leak Rate during Accident	150 gpd
Steam Release from Intact SG to Environment	See Table 6.3-2
SG Iodine-Partition Factor	
Ruptured and Intact SG Steam Release	0.01
Flashed Break Flow	1.0
Primary Coolant Mass	1.19E8 gm
Intact Steam Generator Secondary Mass	84,000 lbm
Ruptured Steam Generator Secondary Mass	84,000 lbm
Offsite Breathing Rates	See Table 6.7-2
Offsite Atmospheric Dispersion Factors	See Table 6.7-2
Control Room Model	See Table 6.7-3
Time to Start Crediting Emergency Control Room HVAC	5 minutes

Table 6.7-12

Assumptions Used LBLOCA Analysis

Source Term	
Core Activity	See Table 6.7-4
Nuclide Parameters	See Table 6.7-1
Activity Release Timing	
Gap Release	First 30 minutes
Fuel Release	1.3 hours (ending at 1.8 hours)
Activity Release from the Fuel	
Noble Gases	5% gap, 95% fuel (100% total)
Iodines	5% gap, 35% fuel (40% total)
Alkali Metals	5% gap, 25% fuel (30% total)
Tellurium Metals	0% gap, 5% fuel (5% total)
Barium, Strontium	0% gap, 2% fuel (2% total)
Noble Metals	0% gap, 0.25% fuel (0.25% total)
Cerium Group	0% gap, 0.05% fuel (0.05% total)
Lanthanides	0% gap, 0.02% fuel (0.02% total)
Iodine Chemical Form in Containment	4.85% elemental, 0.15% organic and 95% particulate
Iodine Chemical Form Released to Atmosphere from ECCS Leakage	97% elemental, 3% organic
Containment Release Path	
Containment, Shield Building, and Auxiliary Building Modeling	See Table 6.7-13
Spray Operation	
Time to Initiate Sprays	0.0 hours
Termination of Spray Injection	0.91 hours
Recirculation Spray	Not credited
Injection Spray Flow Rate	1148 gpm
Spray Fall Height	150 feet
Removal Coefficients	
Elemental Iodine Injection Spray Removal	20.0 hr^{-1}
Particulate Injection Spray Removal	4.5 hr^{-1}
Sedimentation Particulate Removal (after spray termination)	0.1 hr^{-1}

Table 6.7-12 (Cont.)

Assumptions Used for LBLOCA Analysis

ECCS Leakage Release Path	
Credited Sump Volume	315,000 gal
ECCS Leak Rate to Auxiliary Building	12 gal/hr
Airborne Fraction for ECCS Leakage to Auxiliary Building	
0 – 3 hours	10%
> 3 hours	1%
ECCS Leak Rate to RWST	
0 – 24 hours	3 gpm
> 24 hours	1.5 gpm
Airborne Fraction for ECCS Leakage to RWST	1%
ECCS Leakage Plateout in Auxiliary Building	50%
Shield Building and Auxiliary Building Filter Efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Offsite Atmospheric Dispersion Factors	See Table 6.7-2
Offsite Breathing Rates	See Table 6.7-2
Control Room Model	See Table 6.7-3
Time to Start Crediting Emergency Control Room HVAC	2 minutes

Table 6.7-13

Containment, Shield Building, and Auxiliary Building Modeling Used for LBLOCA

Containment Net-Free Volume	1.32E6 (ft^3)
Shield Building Volume	3.74E5 (ft^3)
Shield Building Participation Fraction	0.5
Auxiliary Building Volume	Not modeled, no holdup credited
Containment Leak Rates	
0 – 24 hours	0.5 (weight %/day)
> 24 hours	0.25 (weight %/day)
Containment Leak Path Fractions	
0 – 10 minutes	
Through Shield Building	0.0
Through Auxiliary Building SV	0.1
Direct to Environment	0.9
> 10 minutes	
Through Shield Building	0.89
Through Auxiliary Building SV	0.1
Direct to Environment	0.01
Shield Building Air Flows	
0 – 10 minutes	
Shield Building to Environment	Not applicable
Shield Building Recirculation	Not applicable
10 minutes – 30 minutes	
Shield Building to Environment	6000±10% scfm
Shield Building Recirculation	0.0 scfm
> 30 minutes	
Shield Building to Environment	3100 scfm
Shield Building Recirculation	2300 scfm
Shield Building and Auxiliary Building Filter Efficiencies	
Elemental	90%
Organic	90%
Particulate	99%

Table 6.7-14

Assumptions Used for GDT Rupture Dose Analysis

<u>GDT Inventory (Ci)</u>	
Kr-85m	8.53E1
Kr-85	2.39E3
Kr-87	1.58E1
Kr-88	1.08E2
Xe-131m	5.20E2
Xe-133m	4.76E2
Xe-133	3.85E4
Xe-135m	2.78E1
Xe-135	6.68E2
Xe-138	1.84E0
Nuclide Parameters	See Table 6.7-1
Atmospheric Dispersion Factors	See Table 6.7-2
Breathing Rates	See Table 6.7-2
Control Room Model	See Table 6.7-3 (but with 0.0 scfm unfiltered inleakage)
Time to Start Crediting Emergency Control Room HVAC	30 seconds

Table 6.7-15

Assumptions Used for VCT Rupture Dose Analysis

<u>VCT Inventory (Ci)</u>	
Kr-85m	6.29E1
Kr-85	7.35E2
Kr-87	1.64E1
Kr-88	8.85E1
Xe-131m	2.07E2
Xe-133m	2.21E2
Xe-133	1.62E4
Xe-135m	2.79E1
Xe-135	4.52E2
Xe-138	1.94E0
I-131	8.69E-1
I-132	8.85E-1
I-133	1.30E0
I-134	1.79E-1
I-135	7.09E-1
Duration of Activity Release from Tank	5 minutes
Primary Coolant Noble Gas Activity	1.0% fuel defect level (See Table 6.7-5)
Primary Coolant Initial Iodine Activity	60 μCi/gm of DE I-131 (see Table 6.7 -5)
Letdown Flow Rate	80 gpm +10%
Partitioning of Iodine for Spilled Water	10%
Time to Isolate Letdown Flow	5 minutes
Nuclide Parameters	See Table 6.7-1
Atmospheric Dispersion Factors	See Table 6.7-2
Breathing Rates	See Table 6.7-2
Control Room Model	See Table 6.7-3 (but with 0.0 scfm unfiltered inleakage)
Time to Start Crediting Emergency Control Room HVAC	30 seconds

Table 6.7-16

Assumptions Used for FHA Analysis

Average Assembly Fission Product Activity	See Table 6.7-17
Radial Peaking Factor	1.70
Fuel Rod Gap Fraction	
I-131	8%
Kr-85	10%
Other Iodines and Noble Gases	5%
Fuel Damaged	One assembly
Time after Shutdown	100 hours
Water Depth	23 feet
Overall Pool Iodine Pool Scrubbing Factor	200
Noble Gases Pool Scrubbing Factor	1
Filter Efficiency	No filtration of releases assumed
Isolation of Release	No isolation of releases assumed
Time to Releases All Activity	2 hours
Nuclide Parameters	See Table 6.7-1
Atmospheric Dispersion Factors	See Table 6.7-2
Breathing Rates	See Table 6.7-2
Control Room Model	See Table 6.7-3
Time to Start Crediting Emergency Control Room HVAC	1 minute

Table 6.7-17

Average Fuel Assembly Fission Product Inventory Based on 1782.6 MWt (100.6% of 1772 MWt) and Increased by 6%

	Isotopic Inventory, curies
Iodine	
I-131	2.99E5
I-132	2.53E5
I-133	3.15E4
I-134	0.000E0
I-135	2.25E1
Noble Gases	
Kr-85m	2.22E-2
Kr-85	4.72E3
Kr-87	0.00E0
Kr-88	0.00E0
Xe-131m	4.50E3
Xe-133m	1.06E4
Xe-133	5.75E5
Xe-135m	3.60E0
Xe-135	1.10E3
Xe-138	0.00E0

6.8 Initial Condition Uncertainties and Technical Specification Setpoints

The Kewaunee Nuclear Power Plant (KNPP) Nuclear Steam Supply System (NSSS) accident analyses were evaluated for a 7.4-percent power uprate to a core power of 1772 MWt (current licensed power of 1650-MWt core power). The initial condition uncertainties work supports the 7.4-percent power uprate. The Power Uprating features that were considered are listed below.

The NSSS design parameters that are applicable to the power uprate analyses are shown below:

- Reactor Coolant System (RCS) thermal design flow (TDF) of 178,000 gpm. Some analyses may have assumed a lower bounding TDF.
- Steam generator tube plugging (SGTP) levels between 0 and 10 percent.
- A nominal operating RCS pressure of 2250 psia.
- Where appropriate, mixed cores of FRA/ANP fuel and Westinghouse 422V+ fuel have been addressed, as well as transition effects.
- Reactor coolant average temperature range from 556.3° to 573.0°F for the safety analysis. The reactor coolant average temperature range for some analyses may be wider to bound the above specified T_{avg} range.
- An NSSS power level of 1780 MWt, including a pump heat of 8 MWt (1772-MWt core power).

The safety analyses performed to support the 7.4-percent power uprate used either 1749-MWt core power plus 2-percent uncertainty, or 1772-MWt core power plus 0.6-percent uncertainty.

6.8.1 Initial Condition Uncertainties

Evaluations were performed to determine the initial condition uncertainties for a power uprate for KNPP. The revised thermal design procedure (RTDP) uncertainties were calculated for a total core power uprate of 7.4 percent.

6.8.1.1 Introduction and Background

Since the RTDP methodology is currently under Nuclear Regulatory Commission (NRC) review for KNPP, an evaluation of the various plant parameter uncertainties is required as part of the power uprate evaluation. This evaluation requires a review of pressure, temperature, power, and RCS flow uncertainties that are used in the *Updated Safety Analysis Report* (USAR), Chapter 14. These four parameters and their associated uncertainties are explicitly modeled in various transient analyses. The RCS T_{avg} uncertainty, the calorimetric RCS flow measurement uncertainty, and the calorimetric power measurement uncertainty were recalculated for the NSSS design parameters at power uprate condition. The pressurizer pressure uncertainty is not affected by the uprating but was recalculated with the latest uncertainty methodology.

The power measurement uncertainty evaluated at 1749-MWt core power is based on the use of the venturis installed in each feedwater line to the steam generators. The plant is limited to 1749-MWt core power when using the feedwater venturis to calculate the reactor core power due to the venturi equipment uncertainties. While the KNPP fuel transition analysis was underway, which included a 6.0-percent Power Uprate Program, Westinghouse performed an additional 1.4-percent power uprate (a measurement uncertainty recapture power uprate), based on using ultrasonic flow meters (UFMs) and ultrasonic temperature measurements (UTMs) on the feedwater loops to perform the required calorimetric power measurement. Using the UFMs and UTMs will yield a more accurate determination of feedwater flow and feedwater temperature that are inputs to the calorimetric power calculation. The calorimetric power measurement currently performed for KNPP is accurate to within approximately 2 percent. This 2-percent uncertainty is conservatively applied in the various analyses to demonstrate that the operation of the plant complies with all of the applicable licensing and design criteria. The use of the UFMs, and UTMs with their improved accuracy, will allow an increase on the core power level and a decrease in the power measurement uncertainty to 0.6 percent, while maintaining the validity of the analyses.

6.8.1.2 Input Parameters and Assumptions

The uncertainties were calculated based on KNPP-installed plant instrumentation or special test equipment, and on calibration and calorimetric procedures. The uncertainties associated with the measurement of pressure, temperature, power, and flow are based on the following control and measurement systems:

Pressurizer Pressure	Automatic pressurizer pressure control system
RCS T_{avg}	Automatic T_{avg} rod control system
Reactor Power	Calorimetric power measurement used to normalize power range nuclear instruments
RCS Total Flow	Loop RCS flow measurements based on loop RCS flow channels normalized to a once per fuel cycle calorimetric RCS flow measurement to verify thermal design flow (TDF)

For the 7.4-percent power uprate, information regarding the accuracy of the UFMs and UTMs and the NSSS design parameters shown in Section 2 were used to calculate the RTDP uncertainties, and the power level was adjusted to reflect 1772-MWt core power.

6.8.1.3 Acceptance Criteria

The acceptance criteria for the uncertainties are the values used in the USAR Chapter 14. The RTDP uncertainties must be less than or equal to the uncertainty values used in the USAR Chapter 14 (as shown in Table 6.8-1).

The power measurement uncertainty based on the use of the UFMs and UTMs must be 0.6-percent power or less to allow an additional 1.4-percent calorimetric mini power uprate to 1772-MWt core power.

6.8.1.4 Description of Analyses, Evaluations, and Results

6.8.1.4.1 Methodology

The RTDP methodology is discussed in WCAP-11397-P-A (Reference 1). The uncertainty analysis statistically combines the individual uncertainties using the square root of the sum of the squares (SRSS) method. The analysis includes uncertainties for: the method of measurement (that is, RTDs, transmitters, special test measurements), and the calibration of the instrumentation. The uncertainties for temperature, pressure, power and RCS flow are then used in the development of the reactor fuel departure from nucleate boiling ratio (DNBR) limits. The ΔT reactor trip setpoints are then developed from the new fuel core limits for use in the Technical Specifications. For those transient analyses events using RTDP methodology

(DNB-related transients), the uncertainties are statistically combined with the DNBR correlation uncertainties to obtain the design DNBR limit.

Not all transient analyses events use RTDP methodology. For those transient analyses events (non-DNB-related transients) that do not conform to the RTDP methodology requirements, the standard thermal design procedure (STDP) methodology is employed. The difference between the two methodologies is in the initial conditions used in the transient analysis events and the application of the uncertainties. For the RTDP events (DNB-related transients), the uncertainties are included in the DNBR limit and nominal values are assumed for the initial conditions for RCS pressure, RCS temperature, and power. Minimum measured flow (MMF), that is equivalent to TDF plus a flow uncertainty, is used for RCS flow. For those events using STDP methodology (non DNB-related transients), the uncertainties are directly applied to the nominal values for RCS pressure, RCS temperature, and power to define the initial conditions for the transient analyses events. TDF is used for RCS flow.

Whether positive or negative, uncertainties are applied in a manner that is consistent with the analysis and is in the most conservative direction for a specific event. Analyses that use STDP and those that use RTDP are delineated in the *Reload Transition Safety Report* (RTSR) (Reference 2) and Section 6 of this report.

6.8.1.4.2 Results for the 6.0-Percent Power Uprate

The results of the uncertainty analysis are provided in Table 6.8-1 and WCAP-15591 (References 3 and 4). Also shown are the uncertainties used in the safety analyses. As can be seen in Table 6.8-1, the uncertainties used in the USAR Chapter 14 analysis are actually slightly larger than the calculated values from the RTDP uncertainty analysis. The rationale of using slightly larger values for the uncertainties ensures conservatism in the overall analysis.

6.8.1.4.3 Results for the Additional 1.4-Percent Calorimetric Measurement Uncertaintiy Recapture

Based on the use of the UFMs and UTMs to measure feedwater flow and feedwater temperature respectively, the power measurement uncertainty was calculated to be 0.6 percent (Table 6.8-1). This allows a 1.4-percent measurement uncertainty recapture. This yields a total core power increase from 1749 MWt (from the 6.0-percent power uprating) to 1772 MWt for a total 7.4-percent power uprate.

The RCS flow measurement, the pressurizer pressure, and the T_{avg} uncertainties evaluated at 1749-MWt core power remain applicable for the additional 1.4-percent measurement uncertainty recapture to 1772-MWt core power because there is a minimal impact due to a power increase.

The uncertainties calculated for the 7.4-percent power uprate are applicable over the entire range of 0- to 100-percent power operation (up to 1772-MWt core power).

6.8.1.5 Conclusion

The RTDP uncertainties are shown in Table 6.8-1. The power measurement uncertainty was determined to be 0.6 percent, based on the use of the UFMs and UTMs to measure feedwater flow and feedwater temperature respectively, that allows an additional 1.4-percent measurement uncertainty recapture from 1749-MWt to 1772-MWt core power. The RTDP uncertainties for pressurizer pressure, T_{avg} and RCS flow calculated for the 6.0-percent power uprate bound operation for the additional 1.4-percent measurement uncertainty recapture (up to 1772-MWt core power). The uncertainties are applicable over the entire range of 0- to 100-percent power operation (up to 1772-MWt core power).

6.8.2 Reactor Trip and Engineered Safeguards Technical Specification Setpoints

An evaluation has been performed for the KNPP reactor trip and engineered safeguards Technical Specification setpoints for the 7.4-percent power uprate. The 7.4-percent power uprate includes a 1.4-percent measurement uncertainty recapture power uprate that is based on the use of ultrasonic flow meters (UFMs) and ultrasonic temperature measurements (UTMs). The evaluation determined Technical Specification setpoint applicability and setpoint margins at power uprate operating conditions, and the changes that are required to the Technical Specification setpoint values to ensure adequate protection to the applicable safety analysis limits.

6.8.2.1 Introduction and Background

The KNPP reactor trip and engineered safeguards setpoints in Tables 6.8-2 and 6.8-3 were evaluated for the 7.4-percent power uprate. The evaluation used Westinghouse (Reference 5) and NMC (Reference 6) setpoint analysis methodologies, KNPP fuel transition and power uprate safety analysis limits, and KNPP instrument calibration procedures to assess applicability and margins (Technical Specification and operating margins) for the reactor trip and engineered

protection and engineered safeguards Technical Specification setpoints at the power uprate conditions. The setpoint analysis methodologies used by NMC and Westinghouse in this assessment are consistent with Regulatory Guide 1.105 and conform to ANSI/ISA 67.04, which is a methodology accepted by the NRC per Regulatory Guide 1.105.

6.8.2.2 Input Parameters and Assumptions

Input for this evaluation included KNPP-specific data related to the instrumentation, the associated instrumentation calibration procedures, and the applicable safety analysis limits. The PCWG NSSS design parameters (Section 2) were also used.

6.8.2.3 Acceptance Criteria

For setpoints with defined safety analysis limits, the acceptance criterion is: Margin ≥ 0.

Margin is defined as the difference between the total allowance (TA) and the channel statistical allowance (CSA). TA is the difference between the limiting KNPP USAR Chapter 14 safety analysis limit (that is, within the limits of the calibrated instrument range) and the Technical Specification setpoint (in percent span). CSA is the statistical combination of the instrument channel uncertainty components (in percent span).

For setpoints without defined safety analysis limits, the acceptance criterion is: a low setpoint must be at, or above the bottom end of the instrument range plus the CSA, and a high setpoint must be at, or below the top end of the instrument range minus the CSA .

6.8.2.4 Description of Analyses, Evaluations, and Results

For each setpoint, uncertainty components are determined for the associated instrumentation (transmitters, resistance temperature detector [RTDs], electronics, and associated calibration tolerances), and are combined to determine the CSA.

The uncertainty analysis uses the square root of the sum of squares (SRSS) to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components that are independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, and are then combined by SRSS to determine the CSA.

Table 6.8-2 presents the power uprate reactor trip setpoint values (plant setting value, Technical Specification or COLR value, and safety analysis limits) that ensure KNPP will operate in a manner that will preserve the corresponding safety analysis limits. Table 6.8-2 also provides dynamic time constants for the overtemperature and overpower delta T reactor trip setpoints. These dynamic time constant values that have been determined are used in the safety analyses and will be implemented in the reactor trip plant settings. Note that there are no new time response requirements for these instrument channels.

Table 6.8-3 lists the engineered safeguards setpoint values that ensure that KNPP will operate in a manner that will preserve the corresponding safety analysis limits.

6.8.2.5 Conclusion

The setpoints shown in Tables 6.8-2 and 6.8-3 are applicable for plant operation up to 1772-MWt core power and provide adequate operating, technical specification, and analytical margin for plant operation at the uprated power conditions. The setpoints satisfy the applicable acceptance criteria ensuring adequate protection for the safety analysis limits at power uprate conditions.

6.8.3 References

1. WCAP-11397-P-A, *Revised Thermal Design Procedure*, A. J. Friedland and S. Ray, April 1989.

2. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

3. WCAP-15591, Rev. 0 (Proprietary), WCAP-15592, Rev. 0 (Non-Proprietary) *Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Kewaunee Nuclear Plant (Power Uprate to 1757 MWt – NSSS Power with Feedwater Venturis and 54F Replacement Steam Generators),*), W.H. Moomau, July 2002.

4. WCAP-15591, Rev. 1 (Proprietary), WCAP-15592, Rev. 1 (Non-Proprietary) *Westinghouse Revised Thermal Design Procedures Instrument Uncertainty Methodology, Kewaunee Nuclear Plant (Power Uprate to 1757 MWt – NSSS Power with Feedwater Venturis, or 1780 MWt - NSSS Power with Ultrasonic Feedwater Flow Measurements, and 54F Replacement Steam Generators)*, W.H. Moomau, December 2002.

5. WCAP-15821, *Westinghouse Protection System Setpoint Methodology, Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators*, E. Cervantes, W.H. Moomau, Rev 0 (Proprietary) April 2003.

6. KNPP General Nuclear Procedure (GNP) 04.06.01, *Plant Setpoint Accuracy Calculation Procedure.*

Table 6.8-1

Summary of Initial Condition Uncertainties (applicable for 1749-MWt and 1772-MWt core power)

Parameter	Calculated Uncertainty(1,2)	Uncertainty Allowance Used in Safety Analysis(1,2)
Pressurizer Pressure	[ ]a,c [ ]a,c	±50.0 psi (random) 15.0 psi (bias)
T_{avg}	[ ]a,c [ ]a,c	±6.0°F (random) -1.1°F (bias)
Power (feedwater venturis)	[ ]a,c [ ]a,c	±2.0% RTP (random) 0.4% RTP (bias) (at 1757 MWt - NSSS power)
Power (UFMs and UTMs on feedwater loops)	±0.6% RTP(3) (random)	±0.6% RTP (random) (at 1780 MWt-NSSS power)
RCS Flow (plant computer)	±2.9% flow (random) 0.1% flow (bias)	±4.3% flow (random) 0.1% flow (bias)

Notes:

1. (+) Instrumentation reads higher than the actual parameter.
2. (-) Instrumentation reads lower than the actual parameter.
3. Calculated in Reference 2.

Bracketed [ ]a,c information designates data that is Westinghouse Proprietary, as discussed in Section 1.7 of this report.

Table 6.8-2

Reactor Trip Setpoints

Parameter		Nominal Plant Setting[2]	Tech. Spec. or COLR Value	Safety Analysis Value
High Pressurizer Pressure [2]		2377 psig	<2385 psig	2425 psia 1-sec delay
Low Pressurizer Pressure [2]		1904 psig	>1875 psig	1850 psia 1-sec delay
High Pressurizer Water Level [2]		85%	<90%	100% 1.5-sec delay
OTDT [3]				
Bias	K1	1.1875	<1.20	1.30
Gain on T_{avg} Mismatch	K2	0.015/°F	0.015/°F	0.015/°F
Gain on Pressure	K3	0.00072/psi	0.00072/psi	0.00072/psi
Lead Time Constant on T_{avg} Mismatch	T_1	30 sec	>30 sec	30 sec
Lag Time Constant on T_{avg} Mismatch	T_2	4 sec	<4 sec	4 sec
T'		≤573°F Reference T_{avg} at RTP	≤573.0°F Reference T_{avg} at RTP	≤573.0°F Reference T_{avg} at RTP
P'		2235 psig	2235 psig	2235 psig
fΔI		qt-qb –20.5%+10.5% f(ΔI)= 0 qt-qb > +10.5% slope = 0.96%/% qt-qb < -20.5% slope = 4%/%	qt-qb –22%,+12% f(dI)= 0 qt-qb > +12% slope = 0.96%/% qt-qb < -22% slope = 4%/%	qt-qb –22%,+12% f(dI)= 0 qt-qb > +12% slope = 0.96%/% qt-qb < -22% slope = 4%/%
OPDT [3]				
Bias	K4	1.0825	<1.095	1.16
Gain on T_{avg}	K5	0.0275/°F	0.0275/°F	0.0275/°F
Gain on T_{avg} Mismatch	K6	0.00103/°F	0.00103/°F	0.00103/°F
Rate Lag Time Constant on T_{avg}	T_3	10	>10 sec	10 sec
fΔI		0 for all ΔI	0 for all ΔI	0 for all ΔI
Power Range High Level High Range [2]		105 %	<109%	118% 0.65-sec delay
Power Range High Level Low Range [2]		24.5%	<25%	35% 0.65-sec delay
Power Range Negative Rate [2]		5%Δq/2 sec	<10%Δq/5 sec	N/M[1]

Table 6.8-2 (Cont.) Reactor Trip Setpoints			
Parameter	Nominal Plant Setting[2]	Tech. Spec. or COLR Value	Safety Analysis Limit
Low-Low Steam Generator Water Level (%NRS) [2]	17%	5%	0% 1.5-sec delay
Power Range Positive Rate [2]	5%Δq/2 sec	<15%Δq/5 sec	N/M[1]
Steam Generator Level Low (%NRS) [3] (coincident with SF/FF mismatch)	25.5%	N/A	N/M[1]
SF/FF Mismatch [3] (coincident with SG level low)	0.87E6 lbm/hr	N/A	N/M[1]
Low RCS Flow [3]	93%	>90% loop flow	86.5% 0.75-sec delay
Low Frequency Bus 1, 2 [2]	57.0 cps	>55.0 cps	N/M[1]

Notes :

1. N/M = Not modeled in the safety analysis.
2. Based on NMC evaluations.
3. Based on Westinghouse evaluations.

Table 6.8-3

Engineered Safeguards Setpoints

Parameter	Nominal Plant Setting(2)	Tech. Spec. or COLR Value	Safety Analysis Limit
SI on Low Pressurizer Pressure (2)	. 1830 psig	>1815 psig	1685 psig
SI on Low Steam Line Pressure (2) Lead Time Constant Lag Time Constant	514 psig 12.0 Sec 2.0 Sec	>500 psig >12.0 sec <2.0 sec	480 psig 12.0 sec 2.0 sec
SI on Containment High Pressure (2)	3.70 psig	<4.0 psig	5.0 psig
Steam Line Isolation on High-High Steam Flow and SI (2)	4.35E6 lbm/hr	<4.40E6 lbm/hr	4.47 lbm/hr
Steam Line Isolation on High Steam Flow And Low-Low T_{avg} and SI (3)	0.494 E6 lbm/hr 541.0°F	<0.745E6 lbm/hr >540.0°F	1.75E6 lbm/hr 535°F
Steam Line Isolation on High-High Containment Pressure (2)	16.7 psig	<17 psig	17 psig
Containment Spray (2)	22.4 psig	<23 psig	23 psig
Low T_{avg} (2) Feedwater Regulator Valve Closes	554.5 °F	>554.0°F	554.0°F(4)
Steam Generator Level High (%NRS) (3)	66.5%	N/A	100%

Notes :

1. N/M = Not modeled in the safety analysis.
2. Based on NMC evaluations.
3. Based on Westinghouse evaluations.
4. Assumed in margin-to-trip analysis.

APPENDIX 6A

LOSS OF NORMAL FEEDWATER

LOSS OF NORMAL FEEDWATER

Accident Description

A loss of normal feedwater (from a pipe break, pump failure, or valve malfunction) results in a reduction of the ability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the steam generators, residual heat following reactor trip and reactor coolant pump (RCP) heat would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the Reactor Coolant System (RCS). A significant loss of water from the RCS could conceivably lead to core damage. Controlled shutdown of the reactor and RCS stabilization are also very challenging with a water-solid pressurizer. Since the reactor is tripped well before the steam generator heat transfer capability is reduced, the primary system never approaches a condition where the departure from nucleate boiling ratio (DNBR) limit may be violated.

The following features provide the necessary protection against a loss of normal feedwater:

- Reactor trip on lo-lo water level in two-out-of-three level channels in either steam generator
- Reactor trip on steam flow-feedwater flow mismatch coincident with low water level in either steam generator
- Two motor-driven AFW pumps, which are started on:
  - Lo-lo water level in two-out-of-three level channels in either steam generator
  - Opening of both feedwater pump circuit breakers
  - Any safety injection (SI) signal
  - Loss of offsite power
  - Steam generator anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) lo-lo water level
  - Manual actuation

- One turbine-driven AFW pump, which is started on:
  - — Lo-lo water level in two-out-of-three channels in both steam generators
  - — Loss of voltage on both 4 kV buses
  - — Steam generator AMSAC lo-lo water level
  - — Manual actuation

The Auxiliary Feedwater (AFW) System is started automatically on the signals described above. Below 15 percent of rated thermal power (RTP), select AFW valves (AFW-2A and AFW-2B, and AFW-10A and AFW-10B) can be placed in the closed position (per Technical Specification 3.4.b), thereby precluding automatic delivery of AFW flow to the steam generators. Also below 15 percent of RTP, the control switches for the AFW pumps can be placed in the "pull out" position to prevent filling the steam generators. For these conditions below 15 percent of RTP, operator action to manually establish AFW flow from at least two AFW pumps within 800 seconds (13.3 minutes) after a reactor trip has been determined to be acceptable based on the 100 percent of RTP loss-of-normal-feedwater analysis, in which an 800-second AFW delay has been assumed.

Following a loss of offsite power, the emergency diesel generators supply electrical power to the two motor-driven AFW pumps. The turbine-driven AFW pump is powered via steam flow from the secondary system that exhausts to the atmosphere. All of the AFW pumps are normally aligned to take suction from the condensate storage tank (CST) for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the AFW system is capable of removing the stored energy, residual decay heat and RCP heat following reactor trip. The pressurizer is prevented from becoming water-solid, which could lead to overpressurization of the RCS and a subsequent loss of water from the RCS via a failed-open pressurizer pressure relief or safety valve.

Method of Analysis

The loss-of-normal-feedwater transient is analyzed using the RETRAN computer code. The RETRAN model simulates the RCS, neutron kinetics, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system, and main

steam safety valves (MSSVs). The code computes pertinent plant variables including steam generator mass, pressurizer water volume, and reactor coolant average temperature.

The major assumptions are summarized below.

- The plant is initially operating at 100.6 percent of 1780 MWt NSSS (includes 10 MWt of RCP heat).

- Reactor trip occurs on steam generator lo-lo water level at 0 percent of narrow range span (NRS). Turbine trip occurs coincident with reactor trip.

- A conservative core residual heat generation is assumed, based on the ANS 5.1-1979 decay heat model plus 2 sigma (Reference 1).

- AFW flow from two motor-driven AFW pumps is initiated with flow split equally between the two steam generators (equal split is the limiting case) 800 seconds after the reactor trip on lo-lo steam generator water level. This AFW flow assumption accounts for the limiting single failure that is the loss of the turbine-driven AFW pump. The AFW is modeled as a function of steam generator pressure, and the flow with the first (lowest setpoint) MSSVs open is approximately 170 gpm. The AFW enthalpy is assumed to be 90.8 BTU/lbm (120°F and 1100 psia).

- Secondary system steam relief is achieved through the MSSVs. The MSSVs model includes a +2-percent setpoint tolerance and a 5-psi ramp for the valve to open. Steam relief through the steam generator PORVs and condenser dump valves is assumed to be unavailable.

- The initial reactor coolant vessel average temperature is assumed to be 6°F higher than the nominal full-power value of 573°F because this results in a greater expansion of the RCS water during the transient, thus resulting in a higher pressurizer water level.

- The initial pressurizer pressure is assumed to be 50 psi above the nominal value of 2250 psia. A sensitivity study was performed that demonstrated that a high initial pressurizer pressure is conservative. An additional 0.1-psi uncertainty has been determined to be negligible.

- The initial pressurizer water level is assumed to be 5 percent of span above the nominal value of 48 percent of span, which corresponds to the high nominal full-power vessel average temperature of 573°F. A high initial pressurizer water level is conservative because it minimizes the initial margin to filling the pressurizer water-solid.

- Normal reactor control systems are not assumed to function. However, the pressurizer PORVs, pressurizer heaters, and pressurizer sprays are assumed to operate normally. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient, which would limit the peak pressurizer water volume.

- The initial steam generator water level is assumed to be 7 percent of NRS above the nominal value of 44 percent of NRS. A high initial steam generator water level is conservative because it maximizes the time to reach the steam generator lo-lo water level, thereby maximizing the RCS heatup.

- Reactor trip on steam flow-feedwater flow mismatch coincident with low water level in either steam generator is not credited.

The loss-of-normal-feedwater analysis is performed to demonstrate the adequacy of the Reactor Protection System to trip the reactor and the Engineered Safeguards Features Actuation System (AFW System) to remove long-term decay heat, stored energy, and RCP heat following reactor trip. The actuation of the AFW System prevents excessive heatup or overpressurization of the RCS. As such, the assumptions used in the analysis are designed to maximize the time to reactor trip and to minimize the energy removal capability of the AFW System. These assumptions maximize the possibility of water relief from the RCS by maximizing the expansion of the RCS inventory, as noted in the assumptions listed above.

Results

Figures 6A-1 through 6A-6 show the significant plant responses following a loss of normal feedwater. The calculated sequence of events is listed in Table 6A-1.

Following the reactor and turbine trip from full load, the water level in each steam generator falls due to the reduction of the steam generator void fraction in the tube bundle region, and because of the steam release through the MSSVs, which open to dissipate the RCS stored and generated heat. Eight hundred seconds after the initiation of the lo-lo steam generator water level reactor trip, flow from the two motor-driven AFW pumps is credited, thus reducing the rate of water level decrease in the steam generators.

The capacity of the two motor-driven AFW pumps is sufficient to dissipate core residual heat, stored energy, and RCP heat without water relief through the pressurizer PORVs or safety valves. Figure 6A-4 shows that at no time is there water relief from the pressurizer, as the peak pressurizer water volume is 845.6 ft^3, which is less than the pressurizer volume limit of 1010.1 ft^3. Plant emergency operating procedures may be followed to further cool down the plant. The peak Main Steam System (MSS) pressure is less than 110 percent of the steam generator design pressure. Also, the analysis shows that the RCS overpressurization limit is not challenged during this transient. However, note that the pressurizer sprays and PORVs are assumed to be operable so as to maximize the potential for pressurizer filling. This event is bounded by the loss of external electrical load with respect to peak RCS and MSS pressures.

Conclusions

The results of the loss-of-normal-feedwater analysis show that all applicable acceptance criteria are satisfied. The AFW capacity is sufficient to dissipate core residual heat, stored energy, and RCP heat such that reactor coolant water is not relieved through the pressurizer relief or safety valves.

References

1. ANSI/ANS-5.1-1979, *Decay Heat Power in Light Water Reactors*, August 29, 1979.

Table 6A-1 Sequence of Events – Loss of Normal Feedwater	
Event	**Time (seconds)**
Main Feedwater Flow Stops	20.0
Lo-Lo Steam Generator Water Level Trip Setpoint Reached	53.0
Rods Begin to Drop	54.5
Both Steam Generators Begin to Receive AFW	854.5
Peak Water Level in the Pressurizer Occurs	1165
Core Heat Decreases to AFW Heat Removal Capacity	~1110

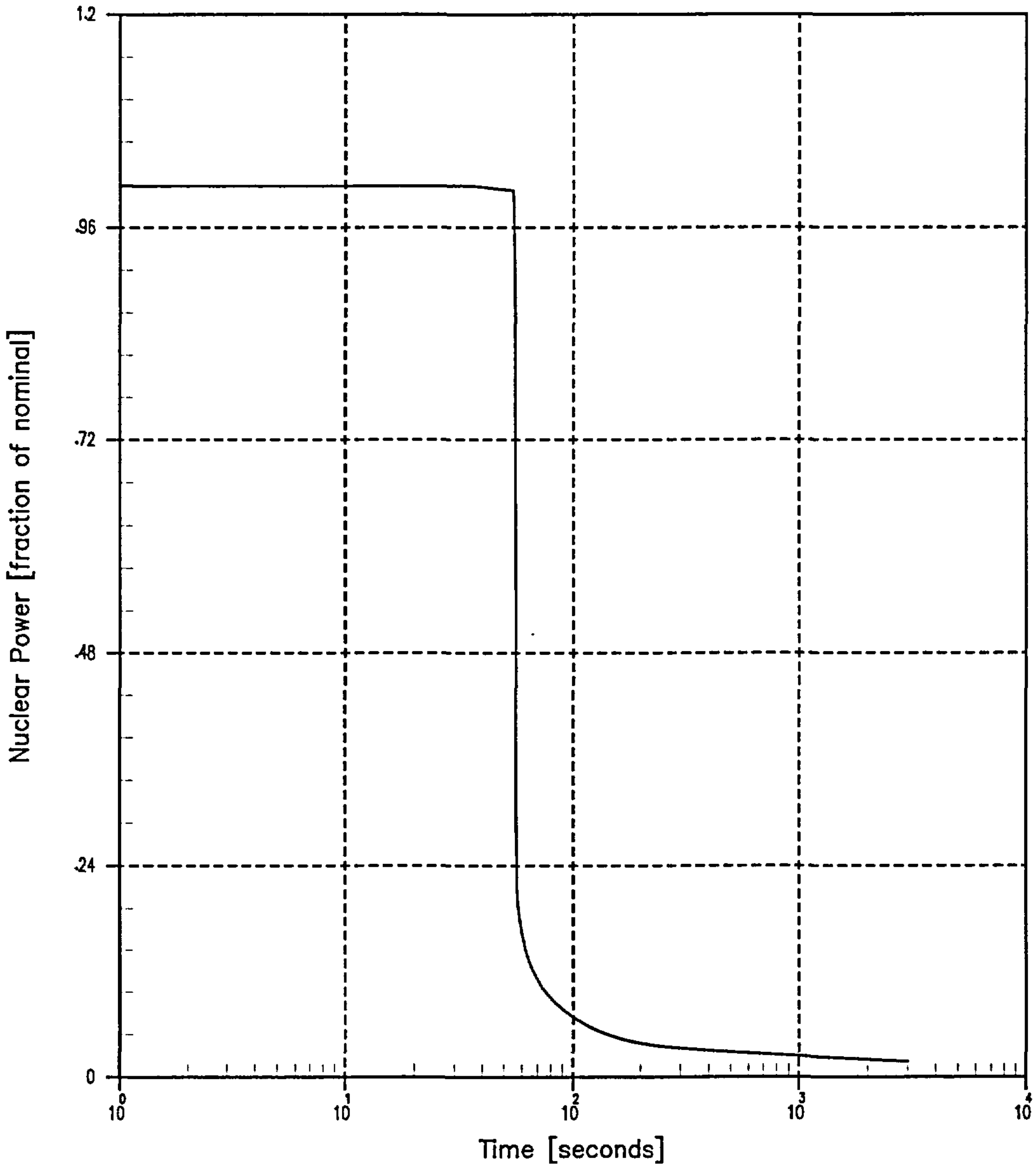

Figure 6A-1

Loss of Normal Feedwater – Nuclear Power

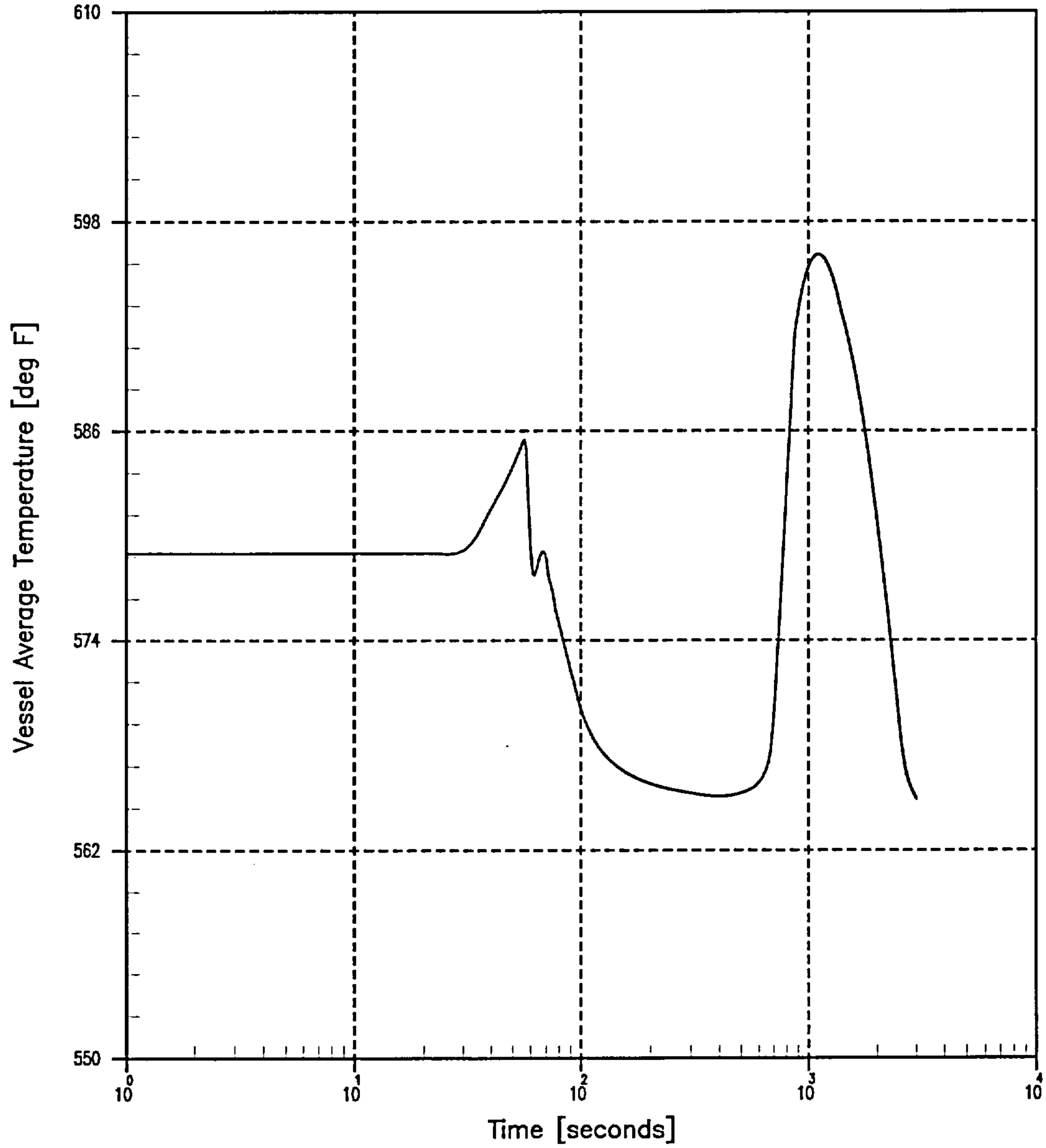

Figure 6A-2

Loss of Normal Feedwater – Vessel Average Temperature

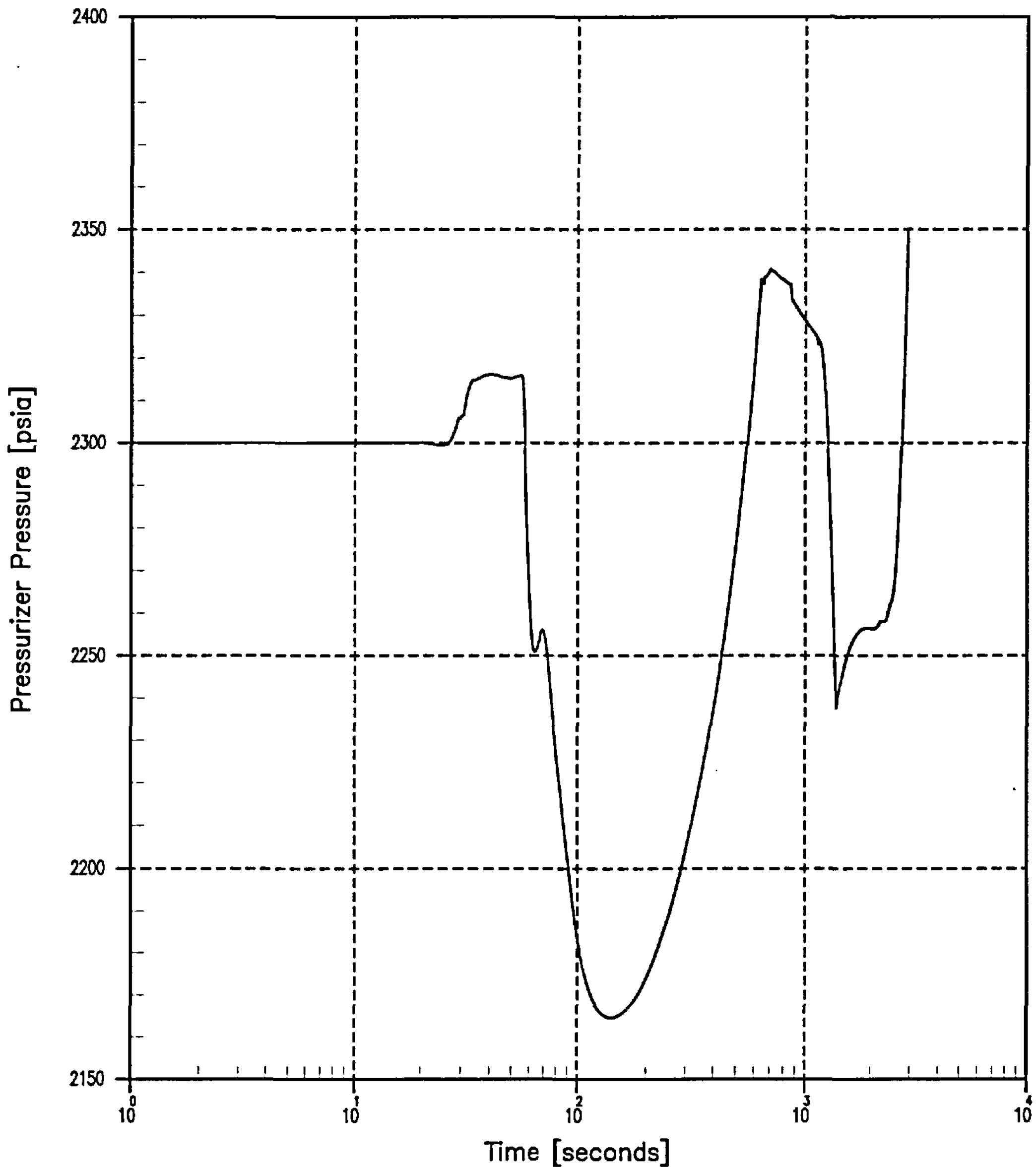

Figure 6A-3

Loss of Normal Feedwater – Pressurizer Pressure

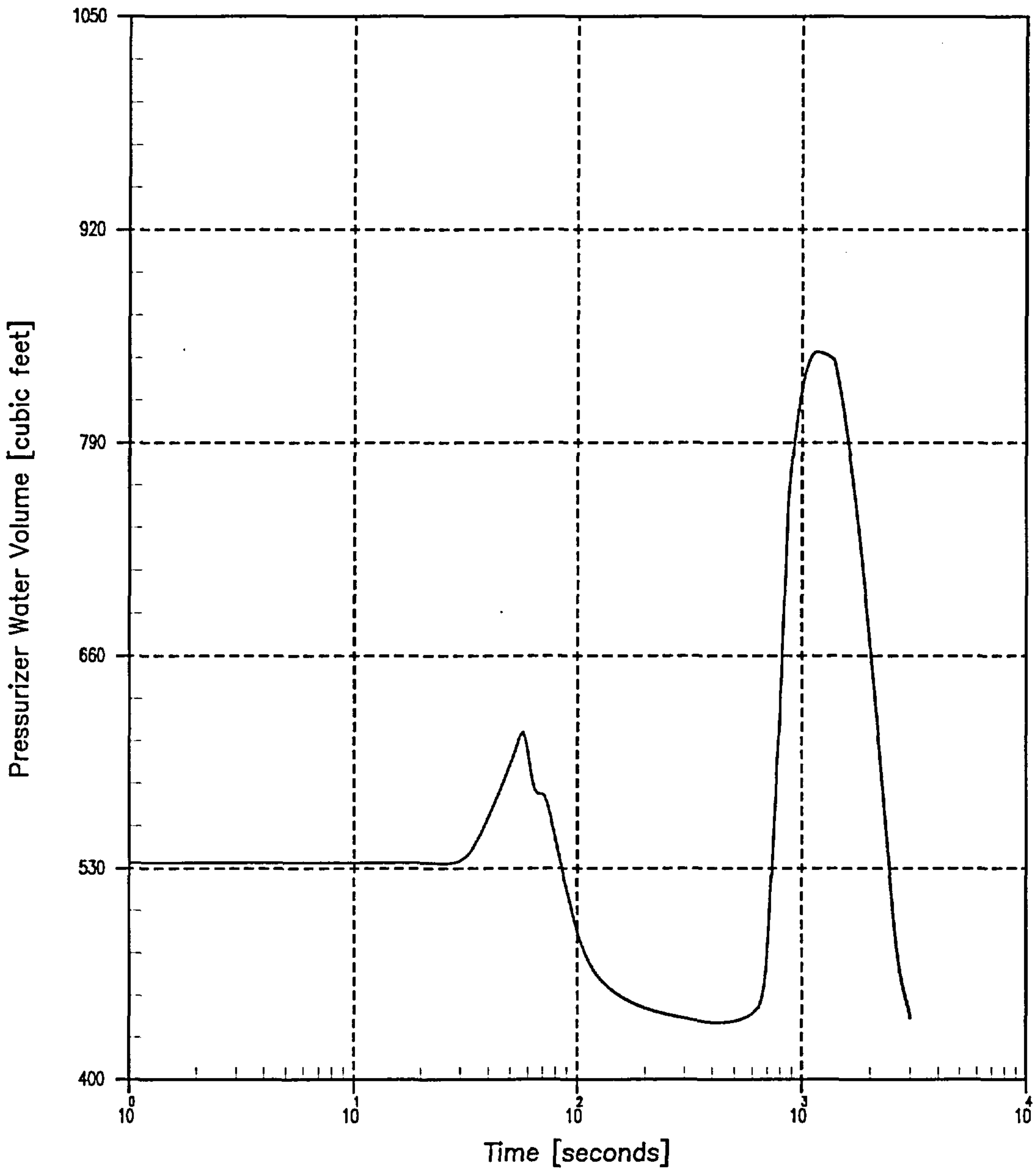

Figure 6A-4

Loss of Normal Feedwater – Pressurizer Water Volume

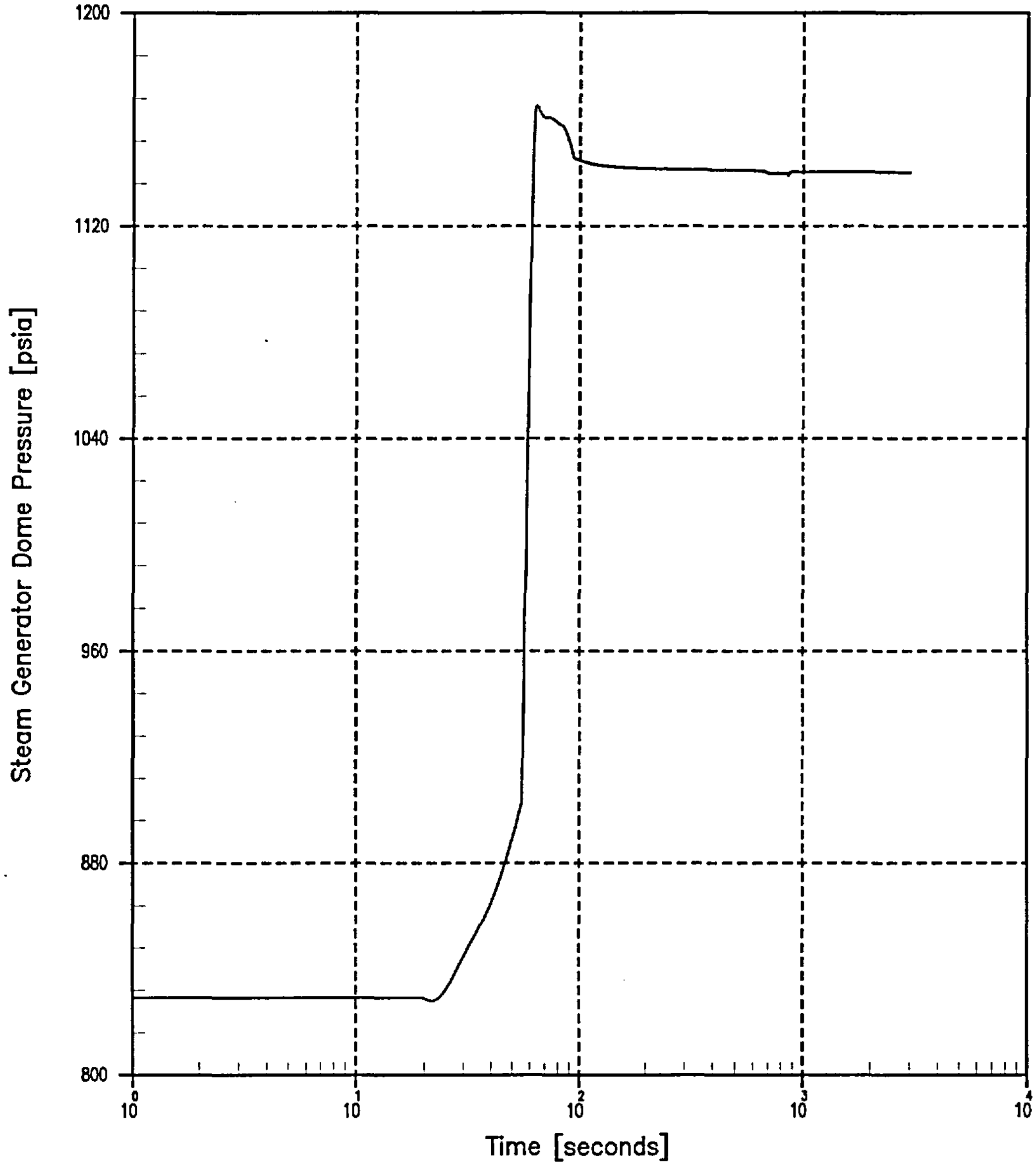

Figure 6A-5

Loss of Normal Feedwater – Steam Generator Pressure

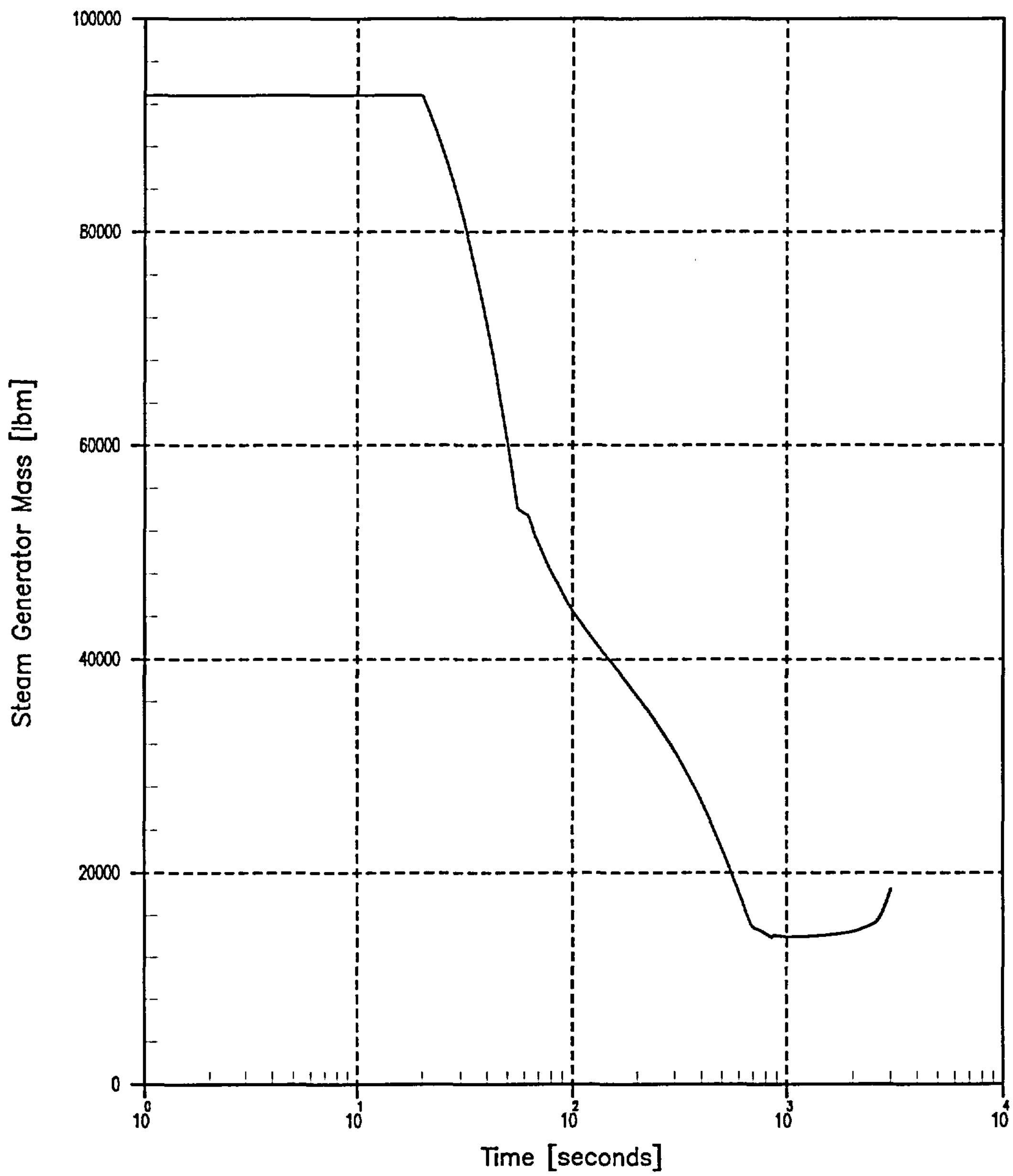

Figure 6A-6

Loss of Normal Feedwater – Steam Generator Mass

7.0 NUCLEAR FUEL

This section summarizes the analyses performed in support of the uprate project in the nuclear fuel and fuel-related areas. The results and conclusions of the analyses are provided in each of the following subsections:

- Core Thermal-Hydraulic Design
- Core Design
- Fuel Rod Design and Performance
- Reactor Internals Heat Generation Rates
- Neutron Fluence
- Radiation Source Terms

7.1 Core Thermal-Hydraulic Design

7.1.1 Introduction and Background

The thermal and hydraulics safety analysis and evaluations performed in support of the Kewaunee Nuclear Power Plant (KNPP) 422V+ fuel (Westinghouse 14x14 422 Vantage + fuel with Performance + Features) *Reload Transition Safety Report* (RTSR) (Reference 1, Section 4) modeled transition core effects and the 7.4-percent power uprate conditions (Table 7.1-1). The analysis encompasses both the transition to and operation with Westinghouse fueled core.

7.1.2 Design Basis and Methodology

The thermal and hydraulic safety analyses performed in the RTSR are based on the Revised Thermal Design Procedure (RTDP) methodology and the WRB-1 departure from nucleate boiling (DNB) correlation (the Standard Thermal Design Procedure [STDP] and the W-3 DNB correlation were used when RTDP and WRB-1 were not applicable). The set of analyses demonstrates that the 95/95 departure from nucleate boiling ratio (DNBR) design basis is met for the 422V+ fuel in the KNPP transition cores and in homogeneous 422V+ cores in operation at the uprated conditions (1772-MWt core power).

The thermal and hydraulic evaluations performed in the RTSR cover the effects of:

- Fuel hydraulic compatibility on the cross flow induced vibration and lift force
- Fuel rod bow effect on DNBR
- Fuel temperature and pressure analyses, core-stored energy calculation
- Transition core effect on DNBR
- Bypass flow

The thermal and hydraulic safety analyses for the Framatome ANP fuel were evaluated by NMC using the NRC-approved models and methods (Reference 4) for KNPP. These models and methods consider the transition core effects and the stretch uprate.

The presence of the 422V+ fuel improves the thermal margins for the Framatome-ANP fuel due to the flow increase experienced by the Framatome-ANP fuel caused by the higher overall loss coefficient in the 422V+ fuel. Note that while the power and T_{ave} increases erode DNBR margin,

the increase in local flow due to the mixed core effects and the decrease in $F_{\Delta H}$ due to the once-burned status of the Framatome ANP fuel both increase DNBR margin.

The thermal and hydraulic safety analyses for the Framatome ANP fuel demonstrates that the DNBR design basis is met for the Framatome ANP fuel in Cycle 26. The detailed documentation of the DNBR design basis is provided in the *Updated Safety Analysis Report* (USAR), completed six months after start-up.[1]

Table 7.1-2 provides an updated DNBR margin summary table for the thimble and typical cells (cell delimited by three fuel rods and one thimble tube and cell delimited by four fuel rods, respectively), applicable to the 422V+ fuel. Compared to the RTSR DNBR margin summary, the following changes were made to accommodate the uprate:

- The design limit DNBR was reduced from 1.24 (RTSR value) to 1.23 by taking into account the latest calculated instrumentation uncertainties (Reference 2).

- An analysis was performed to demonstrate that the rod bow penalty can be reduced from 2.6 to 0 percent for Cycle 26, this was done by conservatively evaluating the FdH burndown DNBR credit during the entire cycle. This is a cycle-specific evaluation. A rod bow penalty of 2.6 percent will be used after Cycle 26.

- The transition core DNBR penalty equation presented in the RTSR was reevaluated using several transition core configurations and following the methodology described in Reference 3. The DNBR transition core penalty for Cycle 26 was calculated to be 2.50 percent (this includes variance for the curve fitting). For Cycle 27 and beyond, a design FdH decrease may also be necessary to generate additional DNBR margin.

There is 2.93-percent DNBR margin to offset any inadvertent event affecting DNBR (flow shortfall, power shape violation, FdH violation). Additional DNBR margin will be available as the core transitions to full 422V+ and the transition core penalty reduces to zero.

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[1] NMC assumes responsibility for this position.

7.1.3 References

1. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

2. WCAP-15591, *Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology - Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators)*, Rev. 1.

3. WCAP-11837, *Extension of Methodology for Calculating Transition Core DNBR Penalties.*

4. U. S. Nuclear Regulatory Commission Letter, *Kewaunee Nuclear Power Plant – Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Rev. 3 (TAC No. MB0306)*, September 10, 2001.

Table 7.1-1

Thermal-Hydraulics Design Parameters

	Analysis Value
Reactor Core Heat Output	1772 MWt
Reactor Core Heat Output	6046 Btu/hr
Heat Generated in Fuel	97.4 %
Nominal (RTDP) Core Pressure	2265 psia
Nominal Pressurizer Pressure	2250 psia
Vessel Flow Rate (RTDP), gpm	186,000
Vessel Average Temperature	573.0°F
Design FdH (include 4% measurement uncertainty)	1.70[1+0.3(1-P)], 422V+
HFP Nominal Conditions	**Analysis Value**
Vessel TDF Rate (including bypass)	67.87 Mlbm/hr 178,000 gpm
Core TDF Rate (excluding bypass)	63.12 Mlbm/hr 165,540 gpm
Core Flow Area	27.1 ft^2
Core Inlet Mass Flux (based on TDF)	2.33 $Mlbm/hr\text{-}ft^2$
Thermal-Hydraulics Design Parameter (based on TDF)	**Analysis Value**
Nominal Vessel / Core Inlet Temperature	539.2°F
Vessel average temperature	573.0°F
Core average temperature	577.1°F
Vessel outlet temperature	606.8°F
Core outlet temperature	611.3°F
Average temperature rise in vessel	67.6°F
Average temperature rise in core	72.1°F
Heat Transfer	**Analysis Value**
Active heat transfer surface area	28,565 ft^2
Average heat flux	206,165 $Btu/hr\text{-}ft^2$
Average linear power	6.85 kW/ft
Peak linear power for normal operation[1]	17.13 kW/ft
Peak linear power for prevention of centerline melt	22.54 kW/ft
Pressure Drop Across Core	**Analysis Value**
Full Core of 422V+[2]	23 psi

Notes:

1. Based on maximum F_Q of 2.50
2. Based on best-estimate reactor flow rate of 98,900 gpm/loop

P = (Thermal Power)/(Rated Thermal Power)

Table 7.1-2 DNBR Margin Summary		
Cell Type	**Typical**	**Thimble**
DNB Correlation[1]	WRB-1	WRB-1
DNBR Correlation Limit	1.17	1.17
DNBR Design Limit[2]	1.23	1.23
DNBR Safety Analysis Limit	1.34	1.34
DNBR Retained Margin	8.21%	8.21%
Rod bow DNBR Penalty	0%	0%
Instrumentation bias DNBR Penalty	-2.78%	-2.63%
Transition core DNBR Penalty[3]	-2.50%	-2.50%
Available DNBR Margin after Uprating	2.93%	3.08%

Notes:

1. W-3 was used for steam line break (SLB) and rod withdrawal from subcritical (RWFS) (below the 1st Mixing Vane Grid), enough DNBR margin was retained to cover rod bow, instrumentation bias and transition core DNBR penalties.
2. Based on the latest calculated instrumentation uncertainties
3. Based on 48 Westinghouse fuel assemblies and 73 FRA/ANP fuel assemblies.

7.2 Core Design

7.2.1 Introduction and Summary

The nuclear design evaluation completed as part of the 422V+ Fuel Upgrade Program (Reference 1) was performed considering an uprated power level of 1772 MWt. The specific values of core safety parameters (power distribution, peaking factors, rod worths, and reactivity parameters) are loading-pattern-dependent and are evaluated each cycle. Standard nuclear design analytical models and methods (References 2, 3, and 4) accurately describe the neutronic behavior of the uprated core.

7.2.2 Design Basis

The specific design basis and its relation to the General Design Criteria (GDC) in the Code of Federal Regulations (CFR) Section 10CFR50, Appendix A, for the power uprate to 1772 MWt are the same as those of the pre-uprate power of 1650 MWt.

7.2.3 Methodology

No changes to the Westinghouse nuclear design philosophy or methods are necessary because of the power uprate to 1772 MWt. The reload design philosophy includes an evaluation of the reload core key safety parameters that comprise the nuclear-design-dependent input to the safety analyses in the reload safety evaluation for each reload cycle (Reference 2). The key safety parameters will be evaluated for each KNPP reload cycle. If one or more of the parameters fall outside the bounds assumed in the reference safety analysis, the affected transients will be re-evaluated using standard methods and the results documented in the Reload Safety Evaluation (RSE) for that cycle.

7.2.4 Design Evaluation – Physics Characteristics and Key Safety Parameters

Three cycles of core models were established that incorporated the uprate to 1772 MWt (Reference 1). Typical loading patterns were developed based on projected energy requirements of approximately 500 effective full-power days (EFPDs) for KNPP. While these loading patterns were not selected to be limiting, they demonstrate that sufficient margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores.

The fuel loading and assembly exposures at beginning of cycle (BOC), and end of cycle (EOC), and assembly power distributions at BOC, middle of cycle (MOC), and EOC are provided for two of the cycles in the RTSR (Reference 1), Figures 3-1 through 3-4. Comparisons of the key core parameters (critical boron concentration, axial offset, hot rod [FNDH], and total peaking factor [FNQ]) versus cycle length are provided in the *Final RTSR* (Reference 1), Figures 3-5 through 3-8, respectively. General trends of the uprated core as a function of burnup are not significantly impacted by the uprate when the same type of loading pattern is used. Cycle-to-cycle loading pattern variations that are addressed by the standard reload methodology may be more significant in shaping the behavior of the core as a function of burnup than the implementation of the power uprate. Normal methods of feed enrichment variation and fresh burnable absorbers can be employed to control peaking factors. Key safety parameter ranges are also provided in the RTSR (Reference 1), Table 3-1.

7.2.5 Nuclear Design Evaluation Conclusions

Key safety parameters were evaluated for KNPP for the uprated condition (1772 MWt). The behavior of the power distributions and peaking factors at the Uprated Power will vary normally from loading pattern to loading pattern, as do typical non-uprated power variations. Typical methods and range of enrichment and burnable absorber variation will be employed in cores designed for the uprated condition to ensure compliance with the peaking factor and reactivity parameter Technical Specifications.

7.2.6 References

1. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

2. WCAP-9273-NP-A, *Westinghouse Reload Safety Evaluation Methodology*, S. L. Davidson, (Ed.), et al., July 1985.

3. WCAP-11596-P-A, *Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores*, T. Q. Nguyen, et al., June 1988.

4. WCAP-10965-P-A, *ANC: A Westinghouse Advanced Nodal Computer Code*, Y. S. Liu, et al., September 1986.

7.3 Fuel Rod Design and Performance

7.3.1 Description of Analyses and Evaluation

The Nuclear Regulatory Commission-approved (NRC–approved) fuel rod performance models (References 1 and 2) and design criteria methods (References 3 and 4) were used to demonstrate that the Kewaunee Nuclear Power Plant (KNPP) 422V+ Fuel Upgrade Program satisfies the NRC Standard Review Plan (Reference 5) at the 7.4-percent power uprate conditions. Each of the fuel rod design criterion listed below have been evaluated at uprated power in the *Reload Transition Safety Report* (RTSR) (Reference 6):

- Rod internal pressure (gap reopening and departure from nucleate boiling (DNB) propagation)
- Cladding stress and strain
- Cladding oxidation and hydriding
- Fuel temperature
- Cladding fatigue
- Clad flattening
- Fuel rod axial growth
- Plenum clad support
- Clad-free standing
- End plug weld integrity

7.3.2 Conclusions

RTSR (Reference 6) evaluations demonstrate that each design criterion can be satisfied through transition cycles to a full core of the 422V+ design. Pending approval of Addendum 1 to WCAP-10125 (Reference 7) by the NRC, subsequent re-evaluation of the stress values will be performed to confirm the proposed clad stress criterion is met.

7.3.3 References

1. WCAP-10581-P-A, *Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations*, R. A. Weiner, et al., August 1988.

2. WCAP-15063-P-A with Errata, *Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)*, J. P. Foster and S. Sidener, July 2000.

3. WCAP-12488-A-P, *Westinghouse Fuel Criteria Evaluation Process*, S. L. Davidson, (Ed.), et al., October 1994.

4. WCAP-13589-P-A, *Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel*, P. J. Kersting et al., March 1995.

5. NUREG-080, *Standard Review Plan - 4.2 - Fuel System Design*, Rev. 2, July 1981.

6. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

7. WCAP-12488, *Revision to Design Criteria*, Addendum 2, April 2002 (was withdrawn at NRC's request and resubmitted as Addendum 1 to WCAP-10125-P-A, December 2002.).

7.4 Reactor Internals Heat Generation Rates

7.4.1 Introduction and Background

The presence of radiation-induced heat generation in reactor internals components, in conjunction with the various reactor coolant fluid temperatures, results in thermal gradients within and between the components. These thermal gradients cause thermal stress and thermal growth that must be considered in the design and analysis of the various components. The primary design considerations are to ensure that thermal growth is consistent with the functional requirements of the components, and to ensure that the applicable ASME Code requirements are satisfied as part of the components evaluation that is in Section 5 of this report. To satisfy these requirements, the reactor internals components must be analyzed with respect to fatigue and maximum allowable stress considerations.

The reactor internals components subjected to significant radiation-induced heat generation are the upper and lower core plates, lower core support, core baffle plates, former plates, core barrel, thermal shield, baffle-former bolts, and barrel-former bolts. However, due to relatively low-heat generation rates in the lower core support and the thermal shield, these components experience little, if any, temperature rise relative to the surrounding reactor coolant.

This section provides a description of the methodology that was used to determine the radiation-induced heat generation rates for the axial core components, that is, the upper and lower core plates; and selected radial reactor internals components, that is, the core baffle, core barrel, and thermal shield, due to the core power uprate to 1772 MWt. Although design-basis neutron exposure data for the reactor internals components are documented in WCAP-9620, Revision 1 (Reference 1), key core power distribution, fuel product, and methodology differences presently exist so that the axial component data reported in WCAP-9620-R1 are non-conservative. However, as demonstrated in the Kewaunee unit-specific analysis performed to support the Uprate Program, the radial component data from WCAP-9620-R1 remains conservative. Key axial components for the Kewaunee Uprate Program were addressed using recently developed baseline upper and lower core plate heating rates applicable to Kewaunee (that is, two-loop design with 1.50-inch core plates).

7.4.2 Description of Analysis and Evaluations, and Input Assumptions

For the core plates, baseline gamma heating rates were determined for both long- and short-term conditions, since the WCAP-9620-R1 (Reference 1) data were no longer deemed applicable for the reactor internals design calculations of these components. Long-term heat generation rates intended to represent time-averaged behavior are used in component fatigue analyses, whereas the short-term results are intended to provide conservative values for use in calculating maximum temperatures and thermal stresses of components. For the long-term heat generation rate evaluation of the core plates, a conservative reactor power level of 2050 MWt was utilized in conjunction with a flat-axial core power distribution, since these parameters significantly influence the core plate gamma heating rates and the aforementioned conditions conservatively bound the Kewaunee Power Uprate Program. For the short-term heat generation rate evaluation of the upper core plate, the reactor power of 2050 MWt continued to be assumed, and a conservative design basis top-peaked axial power distribution from WCAP-9620-R1 (Reference 1) was utilized. Analogous conditions were applied in the short-term heating rate evaluation of the lower core plate; however, in this case, the design-basis bottom-peaked axial power distribution from Reference 1 was employed for conservatism.

For the radial reactor internals components, only a long-term analysis was performed since it was anticipated that the current gamma heating rates for Kewaunee would be bounded by the corresponding data reported in WCAP-9620-R1. (This scenario was hypothesized, since Kewaunee has transitioned to low-leakage loading patterns, whereas an out-in loading pattern was assumed in WCAP-9620-R1. Hence, the long-term case was examined to provide confirmation that the WCAP results remained conservative for the radial components.) Since the long-term radial case of WCAP-9620-R1 was shown to be conservative, the short-term radial case of WCAP-9620-R1 would also remain limiting. The long-term gamma heating rate assessment of the core baffle, barrel, and thermal shield was based on radial power distributions from cycles W1 and W3 of the Kewaunee *Reload Transition Safety Report* (RTSR) (Reference 2) that were individually evaluated at the Kewaunee uprate reactor power level of 1772 MWt. Maximum gamma-heating rate values for the subject radial components were subsequently obtained from the two core reload design cases and were provided for use in the reactor internals evaluations described in Section 5 of this report.

Design-basis heat generation rates applicable to the Kewaunee radial internals were obtained from Appendix G of WCAP-9620-R1. The core power distributions upon which those

calculations were based were derived from statistical studies of 43 independent fuel cycles from 11 two-loop reactors. These power distributions represented an upper tolerance limit for beginning-of-cycle (BOC) and end-of-cycle (EOC) power in the peripheral fuel assemblies, based on a 95-percent probability, with a 95-percent confidence level. Most of the evaluated fuel cycles were based on an out-in fuel loading strategy (fresh fuel on the periphery) that, when combined with the statistical processing of the data, resulted in a design-basis core power distribution that tended to be biased high on the periphery. This high bias on the core periphery was desired by the reactor internals analysts to ensure conservative, but realistic, design calculations for the critical baffle-barrel region of the reactor internals, and explains why the WCAP-9620-R1 radial component heating rate results were expected to bound the corresponding Kewaunee values.

7.4.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to the reactor internals evaluation that is described in Section 5.2 of this report.

7.4.4 Results

The heat generation rate analyses were carried out using the DORT (DOORS 3.1 Code Package, Reference 3) two-dimensional (2-D) discrete ordinates transport code in the forward mode, and the BUGLE-96 cross-section library (Reference 4). This suite of codes has been utilized to support numerous pressure vessel fluence evaluations and is generally accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations, for example, neutron exposure and gamma-ray heating rate evaluations.

Two different coordinate systems were used in the 2-D heating rate analyses (Reference 3) to precisely model the components undergoing evaluation. The core baffle plates were analyzed using an x,y coordinate system, and the core-barrel and thermal-shield heating rates were determined using an r,θ geometric model.

The results of the radiation-induced heat generation rate calculations were provided as inputs for the reactor internals evaluations described in Section 5.2 of this report. The volume-averaged heat generation rates for the core plates and radial reactor internals components that were evaluated as part of this study are summarized in Table 7.4-1. In accordance with WCAP-9620-R1, this table also segregates the core-plate heating rates into two distinct regions.

Region A refers to the cylindrical portion of the core plates that are axially adjacent to the active fuel, and region B refers to the annular portions of the plates that are located radially outboard of the active fuel.

As expected, the revised Kewaunee-zone average gamma heating rates for the core plates tended to be much higher than the corresponding WCAP-9620-R1 data. As a result, the spatial distributions of long-term and short-term heating rates for the upper and lower core plates that are presented in Tables 7.4-2 through 7.4-5, respectively, were also identified for consideration as part of the component evaluation that is described in Section 5 of this report.

Table 7.4-1 also shows that the current Kewaunee-zone average gamma heating rates for the core baffle, core barrel, and thermal shield continue to remain below the conservative radial component heating rates that are reported in WCAP-9620-R1.

7.4.5 Conclusions

The component gamma heating rates applicable to the Kewaunee Nuclear Power Plant (KNPP) for the Fuel Upgrade and Power Uprate Program were determined using input assumptions that were deemed appropriate for this program and the results are summarized in Tables 7.4-1 through 7.4-5. These results are suitable for use in the reactor internals evaluations described in Section 5.2 and represent a valid basis for the transition to, as well as operation with, Westinghouse-fueled cores at 1772-MWt reactor thermal power with respect to reactor internals heat generation rates.

7.4.6 References

1. WCAP-9620, *Reactor Internals Heat Generation Rates and Neutron Fluences,* Rev. 1, A. H. Fero, December 1983.

2. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program,* RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant,* July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

3. RSICC Computer Code Collection CCC-650, *DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System*, August 1996.

4. RSIC Data Library Collection DLC-185, *BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, March 1996.

Table 7.4-1 Reactor Internals Zone Average Gamma Heating Rates		
	Region Average Long-Term Heating Rates (BTU/hr-lbm)	
Location	WCAP-9620-R1 Analysis (Ref. 1)	New Kewaunee Analysis[1]
Baffle Plate 1	757	451
Baffle Plate 2	942	598
Baffle Plate 3	941	599
Baffle Plate 4	841	582
Baffle Plate 5	852	624
Core Barrel	260	143
Thermal Shield	45.2	25.8
	Upper- and Lower-Core Plates Heating Rates (BTU/hr-lbm)	
	WCAP-9620-R1 Analysis (Ref. 1)	New Baseline Analysis
Long-Term Heating Rates		
Upper Core Plate A	30.0	259.8
Upper Core Plate B	7.4	37.7
Lower Core Plate A	276.0	993.2
Lower Core Plate B	65.3	111.6
Short-Term Heating Rates		
Upper Core Plate A	70.9	279.7
Upper Core Plate B	19.5	42.5
Lower Core Plate A	909.0	1592.5
Lower Core Plate B	228.0	199.5

Note:

1. Maximum values from the Kewaunee RTSR cycles W1 or W3 are reported.

Table 7.4-2

Spatial Distribution of Long-Term Gamma Heating Rates (BTU/hr-lbm) in the 1.50-inch Upper Core Plate for the KNPP

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance through Plate (inches) 0.19	0.56	0.94	1.31	Top Surface 1.50
0.59	500.0	470.6	397.0	338.2	290.7	280.0
1.77	490.0	459.9	389.7	333.4	287.9	277.9
2.95	488.4	457.1	386.3	330.4	285.5	275.7
4.13	488.0	457.1	385.2	329.1	284.2	274.5
5.31	486.9	456.0	384.5	328.3	283.4	273.7
6.50	486.0	455.2	383.6	327.6	282.8	273.1
7.68	484.9	454.2	382.9	326.9	282.3	272.5
8.86	484.2	453.5	382.1	326.3	281.7	272.0
10.04	483.4	452.7	381.5	325.7	281.1	271.4
11.22	482.8	452.1	380.9	325.1	280.6	270.9
12.40	481.9	451.3	380.2	324.5	280.0	270.3
13.58	481.0	450.4	379.4	323.8	279.3	269.6
14.76	479.8	449.3	378.4	322.9	278.5	268.8
15.94	478.4	448.0	377.3	321.9	277.6	267.9
17.13	476.7	446.4	375.9	320.8	276.6	266.8
18.31	474.7	444.5	374.3	319.3	275.3	265.6
19.49	472.2	442.2	372.3	317.6	273.7	264.1
20.67	469.3	439.5	370.0	315.6	271.9	262.3
21.85	466.0	436.3	367.3	313.2	269.8	260.2
23.03	462.0	432.5	364.1	310.4	267.4	257.8
24.21	457.3	428.1	360.3	307.2	264.5	255.0
25.39	451.8	422.9	355.8	303.3	261.1	251.7
26.57	445.5	416.9	350.8	298.9	257.3	247.9
27.76	438.1	410.0	344.9	293.9	252.8	243.6
28.94	429.6	402.0	338.1	288.0	247.7	238.7
30.12	419.8	392.8	330.3	281.3	241.9	233.0
31.30	408.7	382.3	321.4	273.8	235.3	226.7
32.48	396.2	370.6	311.5	265.2	228.0	219.6
33.66	382.1	357.4	300.3	255.8	219.8	211.7
34.84	366.3	342.6	288.1	245.2	210.9	203.0
36.02	349.4	326.9	274.7	234.1	201.2	193.7
37.20	331.5	310.0	260.9	222.1	190.8	183.7
38.39	313.1	293.1	246.0	209.1	179.7	173.0
39.57	292.9	273.5	229.3	195.3	167.8	161.7
40.75	269.4	251.7	212.2	180.8	155.6	149.9
42.52	237.3	222.6	187.7	160.1	137.7	132.7
44.09	212.9	199.1	167.5	142.7	122.6	118.1
44.88	199.6	186.6	155.9	132.6	113.9	109.6
45.67	184.9	173.1	145.1	122.9	105.3	101.3
46.46	169.6	159.1	133.5	113.0	96.5	92.8
47.05	158.3	148.4	124.1	105.0	89.5	86.0
47.44	149.0	139.6	116.6	98.5	84.0	80.7
47.83	137.8	129.1	108.1	91.4	78.0	75.0
48.23	125.6	117.6	98.6	83.7	71.7	69.1
48.71	95.5	89.3	76.1	65.1	56.4	55.9
49.27	67.9	64.3	56.4	48.8	43.3	44.1
49.96	58.9	54.2	44.0	37.6	33.5	34.6
50.79	57.6	52.0	38.4	31.0	27.3	28.2
51.61	55.5	50.2	36.0	27.6	23.4	24.2
52.44	50.9	46.1	32.9	24.9	20.6	21.1
53.26	45.1	40.8	28.8	21.7	18.0	18.4
54.09	38.5	34.6	24.2	18.2	15.2	15.6
54.50	32.9	30.0	21.8	16.5	13.7	13.8

Table 7.4-3

Spatial Distribution of Short-Term Gamma Heating Rates (BTU/hr-lbm) in the 1.50-inch Upper Core Plate for the KNPP

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance through Plate (inches) 0.19	0.56	0.94	1.31	Top Surface 1.50
0.59	538.9	507.5	428.9	366.3	315.7	304.3
1.77	528.6	496.2	421.1	361.1	312.8	302.1
2.95	527.0	493.5	417.5	357.9	310.1	299.8
4.13	526.6	493.3	416.3	356.4	308.7	298.4
5.31	525.4	492.2	415.5	355.6	307.8	297.5
6.50	524.5	491.4	414.6	354.9	307.2	296.9
7.68	523.4	490.2	413.8	354.1	306.6	296.2
8.86	522.5	489.4	412.9	353.4	306.0	295.6
10.04	521.5	488.5	412.1	352.7	305.3	294.9
11.22	520.6	487.7	411.3	351.9	304.6	294.3
12.40	519.6	486.7	410.5	351.1	303.9	293.5
13.58	518.4	485.5	409.5	350.3	303.0	292.7
14.76	516.9	484.2	408.3	349.2	302.1	291.8
15.94	515.2	482.6	406.9	348.0	301.0	290.7
17.13	513.2	480.7	405.3	346.6	299.8	289.4
18.31	510.9	478.5	403.4	344.9	298.2	287.9
19.49	508.0	475.8	401.1	342.9	296.5	286.2
20.67	504.7	472.7	398.5	340.6	294.4	284.1
21.85	500.8	469.1	395.3	337.9	292.0	281.8
23.03	496.3	464.8	391.7	334.7	289.1	279.0
24.21	491.0	459.8	387.4	331.0	285.9	275.8
25.39	484.8	454.0	382.4	326.7	282.1	272.1
26.57	477.8	447.3	376.7	321.8	277.7	267.9
27.76	469.6	439.6	370.2	316.1	272.8	263.1
28.94	460.3	430.8	362.7	309.7	267.1	257.6
30.12	449.6	420.7	354.2	302.4	260.8	251.4
31.30	437.6	409.4	344.6	294.1	253.6	244.4
32.48	424.1	396.7	333.9	284.9	245.6	236.7
33.66	409.0	382.7	321.9	274.7	236.8	228.2
34.84	392.2	366.8	308.8	263.5	227.2	219.0
36.02	374.2	350.2	294.6	251.6	216.8	208.9
37.20	355.1	332.2	279.9	238.8	205.6	198.2
38.39	335.6	314.3	264.1	224.9	193.7	186.7
39.57	314.1	293.4	246.1	210.0	181.0	174.5
40.75	289.0	270.0	227.9	194.6	168.0	162.0
42.52	254.9	239.2	201.9	172.6	149.0	143.7
44.09	229.0	214.3	180.4	154.0	132.7	127.9
44.88	214.6	200.8	168.0	143.1	123.3	118.8
45.67	198.6	186.0	156.2	132.7	114.0	109.8
46.46	182.2	171.0	143.7	122.1	104.5	100.6
47.05	170.0	159.4	133.5	113.3	97.0	93.2
47.44	159.8	149.8	125.4	106.3	91.0	87.5
47.83	147.6	138.3	116.1	98.5	84.4	81.4
48.23	134.4	126.0	105.9	90.3	77.7	75.0
48.71	102.8	96.2	82.2	70.6	61.5	60.9
49.27	74.2	70.4	61.8	53.7	47.7	48.6
49.96	65.6	60.5	49.3	42.2	37.5	38.7
50.79	64.7	58.7	43.8	35.5	31.1	32.0
51.61	62.7	57.2	41.5	32.0	27.1	27.7
52.44	58.0	52.9	38.2	29.1	24.0	24.4
53.26	51.5	46.9	33.5	25.4	21.1	21.3
54.09	44.2	39.8	28.1	21.3	17.8	18.1
54.50	37.7	34.4	25.2	19.2	16.0	15.9

Table 7.4-4

Spatial Distribution of Long-Term Gamma Heating Rates (BTU/hr-lbm) in the 1.50-inch Lower Core Plate for the KNPP

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance through Plate (inches) 0.19	0.56	0.94	1.31	Top Surface 1.50
0.59	944.3	977.0	1147.5	1372.6	1667.2	1756.9
1.77	944.1	975.6	1146.6	1371.0	1669.3	1759.0
2.95	944.8	976.2	1147.1	1372.2	1672.1	1761.0
4.13	945.6	977.2	1148.1	1373.3	1673.1	1762.0
5.31	946.4	978.0	1149.0	1374.0	1674.6	1763.5
6.50	947.1	978.8	1149.7	1374.9	1675.9	1764.7
7.68	947.5	979.3	1150.4	1375.9	1676.8	1765.5
8.86	948.0	979.8	1151.2	1376.6	1677.6	1766.3
10.04	948.6	980.5	1151.9	1377.5	1678.7	1767.3
11.22	949.2	981.2	1152.8	1378.6	1680.0	1768.6
12.40	949.9	982.0	1154.0	1380.0	1681.7	1770.3
13.58	950.9	983.2	1155.6	1382.1	1684.1	1772.8
14.76	952.0	984.4	1157.4	1384.4	1686.9	1775.7
15.94	952.8	985.4	1159.0	1386.5	1689.4	1778.4
17.13	953.2	986.0	1160.0	1387.9	1691.2	1780.2
18.31	952.8	985.7	1160.1	1388.2	1691.5	1780.6
19.49	951.4	984.4	1158.9	1386.9	1690.0	1779.1
20.67	948.5	981.5	1155.7	1383.3	1685.6	1774.4
21.85	944.2	977.1	1150.6	1377.0	1677.9	1766.3
23.03	938.7	971.4	1143.9	1368.9	1667.9	1755.7
24.21	932.4	964.8	1136.2	1359.6	1656.4	1743.5
25.39	925.6	957.9	1128.1	1349.8	1644.4	1730.7
26.57	918.7	950.8	1120.1	1340.2	1632.4	1718.0
27.76	911.6	943.8	1112.3	1331.0	1620.8	1705.6
28.94	904.4	936.8	1104.9	1322.6	1610.3	1694.5
30.12	896.3	929.0	1096.9	1313.5	1599.0	1682.6
31.30	886.2	919.1	1086.7	1302.0	1584.6	1667.4
32.48	872.7	905.8	1072.4	1285.7	1564.4	1646.2
33.66	854.3	887.4	1052.0	1262.1	1535.3	1615.7
34.84	829.4	862.0	1023.5	1228.8	1494.3	1572.8
36.02	797.0	828.9	984.6	1183.1	1438.7	1514.5
37.20	755.9	786.3	934.6	1122.8	1365.2	1437.4
38.39	706.2	734.5	872.8	1048.5	1274.1	1341.7
39.57	648.7	674.1	800.4	961.6	1168.5	1230.5
40.75	584.4	606.7	719.6	863.8	1049.6	1105.5
42.52	488.8	506.5	598.0	716.6	870.5	916.9
44.09	407.9	422.0	496.4	593.9	721.3	758.5
44.88	361.5	373.5	437.9	523.6	634.5	667.6
45.67	317.4	327.3	382.9	456.7	552.7	582.0
46.46	276.2	284.1	331.4	394.9	476.6	501.9
47.05	247.8	254.5	296.5	352.8	424.2	446.8
47.44	228.2	233.8	271.8	322.2	386.5	408.2
47.83	208.6	213.1	246.9	292.2	350.6	371.1
48.23	189.1	191.9	222.0	263.6	316.7	335.8
48.71	162.6	161.4	186.1	222.8	275.4	294.9
49.27	137.7	133.9	152.9	185.4	235.6	255.4
49.96	114.6	109.8	124.7	152.9	198.4	216.2
50.79	94.9	90.0	101.5	125.6	164.6	180.8
51.61	79.2	74.5	83.5	103.5	136.8	151.7
52.44	66.0	61.5	68.3	84.7	114.0	127.7
53.26	54.3	50.2	54.7	68.4	94.4	106.8
54.09	43.5	39.7	42.7	53.8	76.3	87.5
54.50	36.8	34.7	37.7	47.3	65.7	74.6

Table 7.4-5

Spatial Distribution of Short-Term Gamma Heating Rates (BTU/hr-lbm) in the 1.50-inch Lower Core Plate for the KNPP

Radial Mesh Midpoint (inches)	Bottom Surface 0.00	Distance Through Plate (inches) 0.19	0.56	0.94	1.31	Top Surface 1.50
0.59	1552.9	1602.3	1858.4	2199.2	2643.5	2776.1
1.77	1553.0	1600.2	1857.2	2197.2	2646.3	2779.0
2.95	1554.4	1601.5	1858.1	2199.0	2650.4	2781.8
4.13	1555.8	1603.0	1859.5	2200.7	2651.8	2783.2
5.31	1557.1	1604.4	1860.8	2201.5	2653.9	2785.2
6.50	1558.1	1605.6	1861.7	2202.8	2655.6	2786.7
7.68	1558.8	1606.4	1862.9	2204.2	2656.8	2787.8
8.86	1559.6	1607.3	1864.0	2205.3	2658.1	2789.1
10.04	1560.4	1608.2	1865.2	2206.6	2659.7	2790.6
11.22	1561.3	1609.2	1866.6	2208.4	2661.7	2792.6
12.40	1562.4	1610.5	1868.4	2210.6	2664.2	2795.1
13.58	1563.7	1612.1	1870.7	2213.5	2667.7	2798.7
14.76	1565.0	1613.7	1873.2	2216.7	2671.6	2802.8
15.94	1566.0	1614.9	1875.3	2219.5	2675.0	2806.3
17.13	1566.1	1615.4	1876.4	2221.2	2677.1	2808.5
18.31	1565.0	1614.5	1876.1	2221.2	2677.0	2808.4
19.49	1562.3	1611.9	1873.6	2218.6	2674.0	2805.3
20.67	1557.4	1607.0	1868.3	2212.5	2666.7	2797.7
21.85	1550.2	1599.6	1860.0	2202.6	2654.7	2785.0
23.03	1541.1	1590.3	1849.3	2189.9	2639.2	2768.6
24.21	1530.7	1579.6	1837.0	2175.2	2621.3	2749.8
25.39	1519.4	1568.1	1824.0	2159.7	2602.4	2729.8
26.57	1507.6	1556.2	1810.7	2144.0	2583.2	2709.5
27.76	1495.2	1543.9	1797.4	2128.4	2564.0	2689.1
28.94	1482.3	1531.3	1784.1	2113.5	2545.5	2669.7
30.12	1467.5	1516.8	1769.4	2097.0	2525.2	2648.2
31.30	1449.1	1498.8	1750.6	2076.1	2499.4	2621.2
32.48	1425.0	1475.0	1725.2	2047.2	2464.3	2584.3
33.66	1393.1	1443.1	1690.1	2006.9	2415.1	2532.9
34.84	1350.9	1400.1	1642.3	1951.4	2347.7	2462.5
36.02	1297.0	1345.0	1578.3	1877.1	2258.1	2368.7
37.20	1229.7	1275.5	1497.6	1780.6	2141.9	2247.3
38.39	1149.3	1191.9	1399.0	1663.2	1999.5	2098.2
39.57	1057.0	1095.2	1284.2	1526.9	1835.7	1926.3
40.75	954.2	987.6	1156.8	1374.3	1652.0	1734.0
42.52	801.8	828.4	966.0	1145.7	1376.7	1445.0
44.09	673.2	694.5	806.8	955.2	1147.6	1202.7
44.88	599.3	617.5	715.1	845.8	1013.9	1063.4
45.67	529.0	544.0	628.7	741.7	887.9	932.0
46.46	463.2	475.3	547.7	645.5	770.4	808.8
47.05	417.7	428.1	492.6	579.6	689.4	723.7
47.44	386.3	395.1	453.7	532.0	631.1	664.3
47.83	354.7	361.8	414.4	485.1	575.6	606.9
48.23	323.1	327.7	375.0	440.4	522.9	552.2
48.71	280.4	279.0	318.5	377.0	459.0	488.8
49.27	240.6	235.2	266.4	319.0	398.1	428.6
49.96	203.7	196.9	222.2	268.8	341.5	368.8
50.79	171.8	164.9	185.4	226.3	290.0	314.8
51.61	145.8	139.1	155.9	190.8	246.2	269.2
52.44	122.9	116.4	129.4	158.8	208.3	229.5
53.26	101.9	95.6	104.5	129.1	173.5	193.1
54.09	81.9	75.8	81.6	101.3	139.9	158.4
54.50	69.7	66.1	71.5	88.5	119.7	134.6

7.5 Neutron Fluence

7.5.1 Introduction and Background

In assessing the state of embrittlement of light-water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of the materials comprising the beltline region of the vessel is required. This exposure evaluation must, in general, include assessments not only at locations of maximum exposure at the inner radius of the vessel, but also as a function of axial, azimuthal, and radial location throughout the vessel wall.

To satisfy the requirements of 10CFR50, Appendix G, for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the Reactor Coolant System (RCS), fast neutron exposure levels must be defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials comprising the beltline region. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper-shelf fracture toughness as specified in 10CFR50, Appendix G. In the determination of values of reference temperature – pressurized thermal shock (RT_{PTS}) for comparison with the applicable pressurized thermal shock screening criterion as defined in 10CFR50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, maximum neutron exposure levels experienced by each of the beltline materials are required. These maximum levels occur at the vessel inner radius.

The methodology used to determine the fast (E > 1.0 MeV) neutron exposure of the Kewaunee pressure vessel derives from the guidance provided in ASTM Standard E853 (Reference 1). The analytical methodology has received regulatory approval as documented in WCAP-14040-NP-A (Reference 2). The Westinghouse methodology has also been documented in WCAP-15557 (Reference 3).

7.5.2 Description of Analysis and Evaluation, and Input Assumptions

A three-dimensional (3-D) assessment of fast (E > 1.0 MeV) neutron exposures for the Kewaunee reactor geometry was made using discrete ordinates transport techniques. The analysis was based on a 2D/1D synthesis of neutron fluxes that were obtained from a series of plant- and cycle-specific forward transport calculations using R-Θ, R-Z, and R spatial mesh. These transport calculations were subsequently compared against dosimetry results obtained

from the in-vessel surveillance capsules withdrawn to date at Kewaunee to demonstrate that the plant-specific analysis meets the 20-percent uncertainty criterion specified in Regulatory Guide 1.190. However, these comparisons only serve to validate the calculational model and are not used in any way to modify the calculational results.

The generalized equation that was used to assess the fast (E > 1.0 MeV) neutron flux in the reactor pressure vessel, which is described in Regulatory Guide 1.190, is given as:

$$\varphi_g(r,\theta,z) = \varphi_g(r,\theta) \times \frac{\varphi_g(r,z)}{\varphi_g(r)}$$

where,

$\phi_g(r,\theta)$	=	The group g transport solution in r,θ geometry for a representative axial plane, that is, at the core midplane.
$\phi_g(r)$ and $\phi_g(r,z)$	=	The one-dimensional (1-D) and two-dimensional (2-D) group g flux solutions whose ratio is used to determine a group-dependent axial shape factor.

The fast (E > 1.0 MeV) neutron exposure calculations were carried out using the DORT (DOORS 3.1 Code Package, Reference 4) discrete ordinates transport code in the forward mode and the BUGLE-96 crosssection library (Reference 5). This suite of codes has been utilized to support numerous pressure vessel fluence evaluations, and is generally accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations, for example, neutron exposure and gamma-ray heating rate evaluations. All calculations were based on an S_{16} order of angular quadrature, and a P_5 expansion of the scattering cross-sections.

The core power distributions used in the plant-specific analysis were taken from the core management nuclear design reports for each of the first 25 operating fuel cycles at Kewaunee. For future projections, cycle W1 from the Kewaunee *Reload Transition Safety Report* (RTSR) (Reference 6) was assumed for the cycle 26 design, and cycle W3 from the Kewaunee RTSR was assumed to be applicable for cycle 27 and beyond. A core power uprate from 1650 MWt to 1772 MWt was also assumed to take place at the onset of cycle 26.

7.5.3 Acceptance Criteria

Adequacy of the modeling is tested by comparing the calculated results against dosimetry measurements from surveillance capsules withdrawn from the plant. As long as these comparisons fall within the ± 20-percent criterion specified in Regulatory Guide 1.190, the calculational results are validated, that is, no specific acceptance criteria apply to the calculated values. However, these calculated results are used as input to reactor vessel analysis that is described in subsections 5.1.1 and 5.1.2 of this report.

7.5.4 Results

The comparisons of calculations with the surveillance capsule dosimetry sets withdrawn to date validated the neutron transport calculations, thus demonstrating that the uncertainty criterion of ± 20 percent (1σ) specified by Regulatory Guide 1.190 is satisfied for the Kewaunee reactor.

The maximum calculated fast (E > 1.0 MeV) neutron fluence and displacement per atom (dpa) exposure values for the Kewaunee pressure vessel are therefore provided in Table 7.5-1. As presented, these data represent the maximum exposure of the pressure vessel clad/base metal interface at azimuthal angles of 0, 15, 30, and 45 degrees relative to the core cardinal axes. The data tabulation includes the plant-specific calculated fluence at the end-of-cycle 25 (EOC 25, the last cycle completed at the Kewaunee plant), and projections for future operation to 24.6, 28, 33, 36, 40, 44, 48, 51, and 54 effective full-power years (EFPYs).

Based on the current NRC position of using the calculated values of neutron fluence to specify the neutron exposure for use in materials damage correlations, the calculated exposure values provided in Table 7.5-1 were provided for use in the materials properties assessments of the Kewaunee pressure vessel at uprated power conditions.

7.5.5 Conclusions

The calculated maximum pressure vessel neutron exposures that are presented in Table 7.5-1 were utilized as inputs to the reactor vessel evaluation described in subsections 5.1.1 and 5.1.2.

7.5.6 References

1. ASTM Standard E853, *Analysis and Interpretation of Light Water Reactor Surveillance Results*, and Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

2. WCAP-14040-NP-A, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves*, January 1996.

3. WCAP-15557, *Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology*, August 2000.

4. RSICC Computer Code Collection CCC-650, DOORS 3.1, *One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System*, August 1996.

5. RSIC Data Library Collection DLC-185, BUGLE-96, *Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, March 1996.

6. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

Table 7.5-1

Summary of Calculated Maximum Pressure Vessel Exposure at the Clad/Base Metal Interface for the KNPP 7.4-Percent Power Uprate Conditions

Neutron Fluence [E > 1.0 MeV]

Cumulative Operating Time [EFPY]	Neutron Fluence [n/cm^2]			
	0.0 Degrees	15.0 Degrees	30.0 Degrees	45.0 Degrees
23.3 (EOC 25)	2.46e+19	1.59e+19	1.29e+19	1.09e+19
24.6	2.59e+19	1.67e+19	1.36e+19	1.16e+19
28	2.98e+19	1.91e+19	1.52e+19	1.30e+19
33	3.56e+19	2.26e+19	1.76e+19	1.51e+19
36	3.90e+19	2.46e+19	1.90e+19	1.64e+19
40	4.36e+19	2.74e+19	2.09e+19	1.81e+19
44	4.82e+19	3.02e+19	2.28e+19	1.98e+19
48	5.28e+19	3.29e+19	2.47e+19	2.15e+19
51	5.63e+19	3.50e+19	2.61e+19	2.28e+19
54	5.97e+19	3.71e+19	2.75e+19	2.41e+19

Iron Atom Displacement

Cumulative Operating Time [EFPY]	Iron Atom Displacements [dpa]			
	0.0 Degrees	15.0 Degrees	30.0 Degrees	45.0 Degrees
23.3 (EOC 25)	4.00e-02	2.67e-02	2.12e-02	1.77e-02
24.6	4.22e-02	2.81e-02	2.23e-02	1.87e-02
28	4.85e-02	3.21e-02	2.49e-02	2.11e-02
33	5.79e-02	3.79e-02	2.88e-02	2.45e-02
36	6.35e-02	4.14e-02	3.12e-02	2.66e-02
40	7.10e-02	4.61e-02	3.43e-02	2.94e-02
44	7.85e-02	5.07e-02	3.74e-02	3.21e-02
48	8.60e-02	5.54e-02	4.05e-02	3.49e-02
51	9.16e-02	5.89e-02	4.28e-02	3.70e-02
54	9.73e-02	6.24e-02	4.51e-02	3.90e-02

7.6 Radiation Source Terms

7.6.1 Introduction and Background

This section describes the input parameters and methodology used in the calculation of radiation source terms applicable to the Uprate Program for the Kewaunee plant. Radiation source terms for several different accident- and normal-operating conditions were determined for the power uprate conditions. These source terms were used as input to dose and balance-of-plant (BOP) analyses. The reanalyzed areas included the following:

- Core inventory and fuel-handling accident fission product activities
- Reactor Coolant System (RCS) sources
- Volume control tank (VCT) sources
- Gas decay tank (GDT) sources
- Tritium generation
- Design basis accident (DBA) sources
- Loss-of-coolant accident (LOCA) DBA environmental qualification (EQ) sources
- LOCA DBA direct and skyshine Control Room dose
- Normal sources
- Decay heat generation
- Residual heat removal (RHR) DBA sources

Each of these source term calculations is discussed in subsequent subsections.

7.6.2 Core Inventory and Fuel-Handling Accident Sources

7.6.2.1 Input Parameters and Assumptions

The assumptions and input parameters used in the determination of the total core inventory are summarized in Tables 7.6-1 and 7.6-2.

7.6.2.2 Description of Analysis

Fuel burnup and fission product production were modeled via the ORIGEN2 code (Reference 1). ORIGEN2 is a versatile point-depletion and radioactive decay code for use in simulating nuclear fuel cycles and calculating the nuclide concentration and characteristics of

materials contained therein. The code considers the transmutation of all isotopes in the material. For the relatively high fluxes in the core region of the reactor, burn-in and burn-out of isotopes can have an important effect. This is particularly true for fuel cycle designs with high-burnup regions. These important effects are modeled in the ORIGEN2 calculations.

For the transition to Westinghouse fuel, the core inventory calculation was performed for the first and third transition cycles. The first transition cycle consists of 77 Framatome/ANP assemblies and 37 Westinghouse assemblies. The third transition cycle consists of all Westinghouse assemblies. Although the core inventory for the first and third transition cycle differed very little, the core inventory for the third transition cycle was greater for most nuclides. For the Kewaunee Uprating Program, the third transition fuel cycle operating at the uprated power conditions was modeled in the ORIGEN2 calculations as the base case. The definition of this transition cycle is provided in Table 7.6-2.

The ORIGEN2 analysis for the Uprating Program modeled a single fuel assembly from each region of the core. Burnup calculations that reflect each of the appropriate power histories were performed, and the total inventory for each region at the end of the transition cycle was then determined by multiplying the individual assembly isotopic inventory by the number of assemblies in the respective regions. Finally, the results for each region of the core were summed to produce the total core inventory.

To consider variations in fuel design and fuel management, inventory sensitivities were used with the third transition cycle inventory and applied to a set of Kewaunee dose analyses. The results showed that a multiplier of 1.06 on the third transition cycle inventory would yield the same or smaller dose for variations of a core average enrichment of 4.5 w/o ± 10 percent, a core mass of 49.1 metric ton unit (MTU) ± 10 percent, and a cycle length of 493.6 effective full-power day (EFPD) ± 10 percent for locked rotor and rod ejection and a multiplier of 1.03 would provide the equivalent margin for LBLOCA.

7.6.2.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to various radiological evaluations.

7.6.2.4 Results

The total core inventory of actinide and fission product activities was provided as input to radiological evaluations that are presented in Section 6.7.

7.6.3 Reactor Coolant System Fission Product Activities

7.6.3.1 Input Parameters and Assumptions

The parameters used in the calculation of the reactor coolant fission product concentrations, including pertinent information concerning the expected coolant cleanup flow rate, are presented in Tables 7.6-2, and 7.6-3. In the RCS activity calculations, fission product escape rate coefficients were used to model a 1-percent level of small cladding defects (that is, 1 percent of the operating fission product inventory in the core is being released to the primary coolant) in all regions of the fuel cycle.

7.6.3.2 Description of Analysis

The fission product inventory in the reactor coolant during operation of the fuel cycle with a 1-percent level of small cladding defects was computed. No credit was taken for fission product removal due to purge of the VCT. Furthermore, in determining the RCS inventory for individual isotopes, the maximum activity occurring at any time during the fuel cycle was documented in each case. Therefore, the total set of fission product concentrations does not represent any particular time during the fuel cycle, but rather, a composite of the maximum activity concentration exhibited by each isotope. This overall approach represents a conservative treatment of the RCS.

For fission products, effects of the following variations were estimated and included conservatively in the calculation of RCS activities:

- Extension of fuel cycle to 525 days at full power
- Enrichment variation by $\pm$10 percent and fuel mass variation by $\pm$10 percent
- Letdown flow decrease by 5 percent
- VCT water volume increase by 10 percent

Corrosion product activities and tritium are standard values.

7.6.3.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to radiological evaluations that are presented in Section 6.7 of this report.

7.6.3.4 Results of Analyses

The RCS fission product and corrosion product specific activities are given in Table 7.6-4 and were provided as input to radiological evaluations that are presented in Section 6.7.

7.6.4 Volume Control Tank Inventory

7.6.4.1 Input Parameters and Assumptions

Radiological inventories given in Table 7.6-4 for the RCS were used as a basis for VCT nuclide concentrations. VCT input parameters used in the nuclide concentration calculation are provided in Table 7.6-3.

7.6.4.2 Description of Analyses

Radiological inventories for the VCT were based on the calculation of RCS and VCT nuclide concentrations with consideration for variations in VCT liquid and gas volumes. Calculations were performed for changes in liquid VCT volume from a nominal 90 ft^3 to 110 ft^3. Maximum values were selected from the nominal and VCT liquid-increase cases.

7.6.4.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to radiological evaluations that are presented in Section 6.7.

7.6.4.4 Results of Analyses

The VCT inventory is given in Table 7.6-5.

7.6.5 Gas Decay Tank Activities

7.6.5.1 Input Parameters and Assumptions

Radiological inventories for the GDTs were based on the RCS-specific activities described in subsection 7.6.3 and the VCT nuclide inventory described in subsection 7.6.4. The input specific activities for the GDT analysis given in Table 7.6-6. For conservatism, the entire calculated GDT inventory, expressed as a volumetric activity, was assumed to be placed in a single GDT.

7.6.5.2 Description of Analyses

GDT activities were calculated by modeling the transfer of the VCT gas contents to the GDT at shutdown, followed by degassing of the RCS at the maximum letdown rate for 3 hours, then followed by purging to the GDT. This cycle of degassing and purging to the GDT is continued for a total of 10 purges, by which time all the nuclides have reached a maximum in the GDT, except for Kr-85.

For a calculation of Kr-85 inventory over 40 years, it is assumed that all the nuclide released to the RCS is eventually deposited in the GDT.

7.6.5.3 Acceptance Criteria

There are no specific acceptance criteria since this is an input to radiological evaluations that are presented in Section 6.7 of this report.

7.6.5.4 Results of Analyses

A summary of the results of this evaluation is given in Table 7.6-7. For Kr-85, the amount at the end of a full cycle is given in this table.

For Kr-85, the GDT inventory as a function of operating time to 40 years is given in Table 7.6-8.

7.6.6 Tritium Generation

7.6.6.1 Input Parameters and Assumptions

Tritium generation is based on the cycle design described in Tables 7.6-2, and 7.6-3; tritium release fractions to the reactor coolant are given in Table 7.6-3.

7.6.6.2 Description of Analyses

Tritium generation is based on a calculation of tritium generation in the active core (fuel rods and coolant water) from ternary fissions, soluble boron and lithium in the coolant, and deuterium reactions in the coolant during normal operation.

The tritium generation calculations use the reactor power level; a set of groupwise neutron fluxes; groupwise neutron reaction cross sections; and water masses in the active core region, to predict the tritium generation. The design value of release of tritium from ternary fissions to the coolant is 10 percent of generation; the expected value is 2 percent.

7.6.6.3 Acceptance Criteria

The results of the tritium source analysis are used to evaluate plant tritium generation and release. There are no acceptance criteria for these stand-alone calculations.

7.6.6.4 Results of Analyses

The calculated tritium generation and release to the reactor coolant is provided for evaluation of plant tritium releases. A summary of the results of this tritium generation and release analysis is given in Table 7.6-9.

7.6.7 Design Basis Accident Sources

7.6.7.1 Input Parameters and Assumptions

The calculation follows the method of Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, (Reference 2) for calculating the LOCA DBA source. Input parameters used in the source calculation are

provided in Tables 7.6-1 and 7.6-2, which provide the pre-accident core inventory basis for the ORIGEN2 calculation.

7.6.7.2 Description of Analyses

The method of calculation employs the ORIGEN2 code to generate the core inventory of actinides and fission products and to calculate the energy released according to selected inputs. The fractions of core inventory released, as a function of time, are based on Regulatory Guide 1.183.

7.6.7.3 Acceptance Criteria

The calculation provides radiation sources to be used as input to safety analysis and shielding calculations. As such, there are no specific criteria for these calculated results.

7.6.7.4 Results of Analyses

Table 7.6-10 gives the DBA gamma source in units of MeV/sec. Figure 7.6-1 gives the gamma dose rate in units of MeV/sec. Figure 7.6-2 gives the gamma dose in units of MeV.

7.6.8 Loss-of-Coolant-Accident Design Basis Accident Direct and Skyshine Control Room Dose

7.6.8.1 Input Parameters and Assumptions

The radiation source is taken from subsection 7.6.7of this report, and is based on the alternate source term (AST) from Section 6.7 of this report. Additional parameters include the reactor containment vessel and containment shield building dimensions, the Control Room location relative to the Reactor Containment Building, and the location and dimensions of selected Auxiliary Building walls and floors.

7.6.8.2 Description of Analyses

The gamma radiation penetrating the Control Room from skyshine (air scatter) and the gamma radiation going directly from the containment into the Control Room is calculated, using a Monte Carlo technique as performed by the Monte Carlo N-Particle (MCNP) computer code. In the calculation, the radiation source is placed in the containment vessel. Gamma rays escape

containment and the concrete shield and interact with air. Some gamma rays will go directly to the Control Room; others will be scattered from air (or ground) to the Control Room.

Additional shielding provided by the Auxiliary Building is evaluated and combined with MCNP results to demonstrate a low dose to the Control Room.

7.6.8.3 Acceptance Criteria

The calculation provides a radiation source to be used as input to the LOCA Control Room dose. As such, there are no specific criteria for this portion of calculated results.

7.6.8.4 Results of Analyses

The 1-month calculated dose in the Control Room is 0.034 rem. Applying an additional 6 percent for fuel management variations gives a Control Room dose of 0.036 rem.

The Control Room dose as a function of time after the accident is given in Table 7.6-11. Dose rate and dose are illustrated in Figure 7.6-3.

7.6.9 Normal Sources

7.6.9.1 Input Parameters and Assumptions

This calculation uses the ANSI/ANS-18.1-1984 (Reference 3) specifications and formulations for calculating the radionuclide activity in the fluid streams of a light-water reactor (LWR) nuclear plant. The use of this standard is consistent with the methodology considered by the NRC in its review of expected plant radioactive effluents for all LWR plants. The ANSI/ ANS-18.1-1984 data is scaled by thermal power level and other pertinent parameters as outlined in the ANS Standard. Specific inputs related to the Kewaunee plant are given in Table 7.6-12.

The numerical values given in the ANSI/ANS-18.1-1984 standard are based on available data from operating plants that use Zircaloy-clad, uranium-dioxide fuel. Normal sources for the Kewaunee plant are established by appropriate scaling and calculations to provide source values specific to the Kewaunee nuclear plant.

7.6.9.2 Description of Analyses

The normal source calculation of noble gas distributions in the RCS and steam generator steam or water are calculated based on the assumption of either: no purge of the VCT per ANSI/ANS-18.1-1984; or a VCT purge of 0.7 scf/minute.

7.6.9.3 Acceptance Criteria

The results of the normal plant operation source calculations are used to establish the long-term concentrations of principal radionuclides in the fluid streams of the plant for subsequent use in estimating the expected release of radioactive materials from various effluent streams. The fluid streams of the plant are the reactor coolant water and the steam generator water and steam. There are no specific acceptance criteria for these stand-alone calculations.

7.6.9.4 Results of Analyses

Table 7.6-13 gives the normal sources based on ANSI/ANS-18.1-1984 for the Kewaunee plant.

7.6.10 Decay Heat Generation

7.6.10.1 Input Parameters and Assumptions

Fuel burnup with fission product and actinide production were modeled with the ORIGEN2 code (Reference 1). The assumptions and input parameters used in ORIGEN2 are those used in the determination of the total core inventory, which are summarized in Tables 7.6-1 and 7.6-2.

Regulatory Guide 3.54 (Reference 4) was used for the decay heat of assembly activation products. Available calculation results for the decay heat of activated reactor internals is included in the core "total" decay heat.

7.6.10.2.1 Description of Analyses

A core total decay heat curve was developed for up to 720 hours after shutdown. Decay heat contributions are included from:

- Fission products
- Actinides

- Assembly activation products
- Reactor internals activation products
- Fuel management variations

Contributions to decay heat from reactor internals were calculated assuming extended reactor irradiation (to 54 effective full-power years [EFPYs]) to provide decay heat from activated stainless steel components.

Sensitivities of fission product and actinide decay heat were determined to be a 2.2 percent increase in core decay heat for increases in burnup and enrichment. Since the multiplier of 1.06 is used in the results, there is approximately 6 percent - 2.2 percent = 3.8 percent conservatism remaining in results reported. A multiplier of 1.06 was used to accommodate the effect of fuel management variations on dose calculations.

The decay heat for a single assembly was developed for a Kewaunee assembly achieving a burnup of 62,000 MWD/MTU, which should represent a high-decay-heat assembly. The decay heat projected for 60 years included decay heat contributions from:

- Fission products
- Actinides
- Assembly activation products

The decay heat for this assembly does not have specific additions for conservatism, but the following factors provide a conservative decay heat:

- The power level used throughout the 62,000 MWD/MTU is the core average assembly power at core power of 1782.6 MWt. Fuel management characteristics will cause the power in the third (or fourth) cycle to be below the average assembly power. This would lower decay heat at discharge.

- A burnup of 62,000 MWD/MTU for an assembly is higher than anticipated.

After the reactor is tripped, fissioning of considerable magnitude continues due to delayed neutrons for a brief time, but rapidly diminishes (after about 100 seconds) to an insignificant level relative to the heat produced by fission product and actinide decay. The items of interest for this calculation are reactor cooldown and spent fuel heat loads, for which the contribution of

fission by delayed neutrons is negligible. Therefore, residual heat due to delayed neutron fissioning is not accounted for in these analyses.

7.6.10.3 Acceptance Criteria

The selection of source terms from multiple cases is made to provide a bounding decay heat curve for use in engineering calculations and plant procedures. There are no specific acceptance criterion that apply to the determination of these sources. Rather, the acceptance criterion is imposed on the results of subsequent calculations that use decay heat as input.

7.6.10.4.1 Results of Analyses

Core decay heat from shutdown to 1 month is given in Table 7.6-14. Figure 7.6-4 illustrates the results. This core decay heat may be used in cool-down analysis and plant procedures, since core internals and assembly activation products are explicitly included.

Assembly decay heat from shutdown to 60 years for one assembly with enrichment of 4.5 w/o and a burnup of 62,000 MWD/MTU is given in Table 7.6-15. Figure 7.6-5 illustrates the results for the period 0 to 180 days of decay, and for 1 year to 60 years decay.

7.6.11 References

1. RSIC Computer Code Collection CCC-371, ORIGEN2.1: *Isotope Generation and Depletion Code – Matrix Exponential Method*, February 1996.

2. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000.

3. ANSI/ANS-18.1-1984, *American National Standard Radioactive Source Term Normal Operation of Light Water Reactors*, American Nuclear Society, LaGrange Park, Illinois, December 31, 1984.

4. Regulatory Guide 3.54, *Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation*, September 1984.

Table 7.6-1 Input Parameters for Core Inventory Calculations	
Parameter	**Value**
Core Thermal Power (MWt)	1782.6 (1772*1.006)
Fuel Assembly Type	422V+
Uranium Mass (MTU)	40.065
Length (MWD/MTU)	17827
Loading Pattern	See Table 7.6-2
Uranium Enrichments (wt % U-235)	Region R30A 4.383 Region R30B 4.716 Region R29A 4.364 Region R29B 4.709 Region R28CT 4.362 Region R28T 4.709

Table 7.6-2
Input Parameters for Third Transition Fuel Cycle

Region	No. of Assemblies	EOC Burnup (MWD/MTU)	Average Relative Power
Feed Region R30A	36	22850	1.28
Feed Region R30B	8	19700	1.11
1 x Burned Region R29A	21	44926	1.15
1 x Burned Region R29B	24	39725	1.06
2 x Burned Region R28CT	8	51947	0.73
2 x Burned Region R28T	24	50506	0.45

Table 7.6-3

Input Parameters for RCS Activity and Inventory Calculations

Parameter	Value
Core Thermal Power (MWt)	1782.6 (1772*1.006) 1772 for tritium generation
Fuel Assembly Type	422V+
Uranium Enrichment (wt % U-235)	Region R30A 4.383 Region R30B 4.716 Region R29A 4.364 Region R29B 4.709 Region R28CT 4.362 Region R28T 4.709
Uranium Mass (MTU)	40.065
Cycle Length (MWD/MTU)	17827
Initial Boron Concentration (ppm)	1520
Mixed Bed Demineralizer Resin Volume (ft^3)	30
Cation Bed Demineralizer Resin Volume (ft^3)	12
Failed Fuel Fraction (%)	1.0
Reactor Coolant Mass (lbm)	2.627×10^5
Purification System Flow Rate, Normal (gpm)	40
Purification System Flow Rate, Maximum (gpm)	80
Volume Control Tank total Volume (ft^3)	220
Volume Control Tank Temperature (°F)	140
Tritium Release Fraction from Fuel Rods	
Design Bases	0.1
Expected	0.02
RCS Lithium Concentration (ppm)	3

Table 7.6-4

Reactor Coolant Fission and Corrosion Product-Specific Activities

Nuclide	Activity μCi/g	Nuclide	Activity μCi/g	Nuclide	Activity μCi/g
Kr-83m	4.31E-01	Rb-86	3.16E-02	Zr-95	6.53E-04
Kr-85m	1.73E+00	Rb-88	4.09E+00	Nb-95	6.55E-04
Kr-85	8.60E+00	Rb-89	1.87E-01	Mo-99	7.60E-01
Kr-87	1.13E+00	Cs-134	2.86E+00	Tc-99m	7.07E-01
Kr-88	3.28E+00	Cs-136	3.22E+00	Ru-103	5.59E-04
Kr-89	9.26E-02	Cs-137	2.16E+00	Ru-106	1.89E-04
Xe-131m	3.04E+00	Ba-137m	2.04E+00	Rh-103m	5.54E-04
Xe-133m	3.44E+00	Cs-138	9.63E-01	Rh-106	1.89E-04
Xe-133	2.42E+02	H-3	3.50E+00	Ag-110m	1.54E-03
Xe-135m	5.01E-01	Cr-51	5.40E-03	Te-125m	6.92E-04
Xe-135	8.69E+00	Mn-54	4.00E-04	Te-127m	3.13E-03
Xe-137	1.73E-01	Fe-55	2.10E-03	Te-127	1.32E-02
Xe-138	6.28E-01	Fe-59	5.10E-04	Te-129m	1.07E-02
Br-83	9.11E-02	Co-58	1.40E-02	Te-129	1.38E-02
Br-84	4.51E-02	Co-60	1.30E-03	Te-131m	2.50E-02
Br-85	5.25E-03	Sr-89	4.17E-03	Te-131	1.35E-02
I-127 (*)	1.05E-10	Sr-90	2.07E-04	Te-132	2.94E-01
I-129	6.25E-08	Sr-91	5.57E-03	Te-134	2.87E-02
I-130	3.44E-02	Sr-92	1.24E-03	Ba-140	4.16E-03
I-131	2.84E+00	Y-90	5.82E-05	La-140	1.41E-03
I-132	2.89E+00	Y-91m	3.01E-03	Ce-141	6.36E-04
I-133	4.24E+00	Y-91	5.61E-04	Ce-143	4.86E-04
I-134	5.86E-01	Y-92	1.08E-03	Ce-144	4.81E-04
I-135	2.32E+00	Y-93	3.58E-04	Pr-143	6.11E-04
				Pr-144	4.81E-04
(*) g/g of water					

Note:

The reactor coolant activities are based on RCS mass = 1.19 x 10^8 grams.

Table 7.6-5

Nuclide Inventories for Noble Gases and Iodine in the VCT (total of gas and liquid phases)

VCT Isotope	Inventory (curies)
Kr-85m	6.29E+01
Kr-85	7.35E+02
Kr-87	1.64E+01
Kr-88	8.85E+01
Xe-131m	2.07E+02
Xe-133m	2.21E+02
Xe-133	1.62E+04
Xe-135m	2.79E+01
Xe-135	4.52E+02
Xe-138	1.94E+00
I-131	8.69E-01
I-132	8.85E-01
I-133	1.30E+00
I-134	1.79E-01
I-135	7.09E-01

Table 7.6-6 RCS and VCT Activities Input to the GDT Calculation		
Nuclide	**RCS uCi/g**	**VCT uCi/cc**
Kr-85m	1.73E+00	1.65E+01
Kr-85	8.60E+00	1.93E+02
Kr-87	1.13E+00	4.28E+00
Kr-88	3.28E+00	2.31E+01
Xe-131m	3.04E+00	5.37E+01
Xe-133m	3.44E+00	5.73E+01
Xe-133	2.42E+02	4.22E+03
Xe-135m	5.01E-01	7.56E+00
Xe-135	8.69E+00	1.18E+02
Xe-138	6.28E-01	5.00E-01

Table 7.6-7 GDT Sources after Shutdown	
GDT Isotope	**Inventory (curies)**
Kr-85[(1)]	2.39E+03
Kr-85m	8.53E+01
Kr-87	1.58E+01
Kr-88	1.08E+02
Xe-131m	5.20E+02
Xe-133	3.85E+04
Xe-133m	4.76E+02
Xe-135	6.68E+02
Xe-135m	2.78E+01
Xe-138	1.84E+00

Note:

1. Inventory at the end of one cycle.

Table 7.6-8 Buildup of GDT Kr-85 Over 40 Years of Operation	
Time (years)	**Inventory (curies)**
1	1.79E+03
2	3.47E+03
3	5.04E+03
4	6.51E+03
5	7.90E+03
6	9.19E+03
7	1.04E+04
8	1.15E+04
9	1.26E+04
10	1.36E+04
15	1.77E+04
20	2.07E+04
25	2.29E+04
30	2.45E+04
35	2.56E+04
40	2.64E+04

Table 7.6-9 Reactor Coolant Tritium Activity (curies per cycle)			
Tritium Source	Total Produced (curies)	Released to the Coolant	
		Design Value (curies)	Expected Value (curies)
Ternary Fissions	8750	875	175
Soluble Poison Boron	355	355	355
Li-7 Reaction	13.1	13.1	13.1
Li-6 Reaction	89.5	89.5	89.5
Deuterium Reaction	1.7	1.7	1.7
Total Cycle	9209.3	1334.3	634.3

Table 7.6-10

Radiation Sources Released to the Containment Following the DBA - MEV/sec[1]

Gamma Energy (MeV/photon)	Decay Time 0 (min)	10 (min)	40 (min)	59.5 (min)	1.32 (hours)	1.97 (hours)	3.97 (hours)	10 (hours)	24 (hours)	7 (days)	30 (days)
0.01	0	1.35E+14	1.85E+14	1.96E+15	2.71E+15	3.36E+15	2.59E+15	1.64E+15	1.25E+15	5.43E+14	4.24E+13
0.025	0	2.94E+14	3.91E+14	3.34E+15	5.07E+15	8.14E+15	6.35E+15	5.46E+15	4.64E+15	1.69E+15	2.57E+14
0.0375	0	1.90E+15	3.06E+15	3.37E+16	4.83E+16	6.23E+16	6.13E+16	5.94E+16	5.52E+16	2.59E+16	1.51E+15
0.0575	0	2.78E+13	4.39E+13	3.55E+14	6.54E+14	1.23E+15	1.21E+15	1.15E+15	1.04E+15	4.01E+14	5.64E+13
0.085	0	3.49E+15	5.83E+15	6.55E+16	9.39E+16	1.20E+17	1.19E+17	1.16E+17	1.10E+17	5.23E+16	2.63E+15
0.125	0	1.08E+15	1.27E+15	5.34E+15	7.64E+15	9.59E+15	4.91E+15	3.35E+15	2.73E+15	6.68E+14	9.68E+13
0.225	0	2.32E+16	2.92E+16	2.60E+17	3.69E+17	5.02E+17	4.43E+17	3.28E+17	1.68E+17	1.68E+16	1.78E+15
0.375	0	4.45E+16	5.69E+16	2.36E+17	3.23E+17	4.20E+17	3.19E+17	2.54E+17	2.23E+17	1.31E+17	1.95E+16
0.575	0	2.25E+17	3.21E+17	9.92E+17	1.48E+18	2.26E+18	1.57E+18	1.00E+18	6.53E+17	1.91E+17	1.26E+17
0.85	0	3.39E+17	4.23E+17	1.33E+18	1.89E+18	2.43E+18	1.21E+18	4.92E+17	3.04E+17	1.52E+17	9.18E+16
1.25	0	3.97E+17	4.61E+17	1.23E+18	1.68E+18	2.13E+18	1.34E+18	6.68E+17	2.32E+17	4.09E+16	1.48E+16
1.75	0	1.40E+17	1.68E+17	7.02E+17	1.00E+18	1.27E+18	8.45E+17	3.24E+17	8.99E+16	7.48E+16	2.31E+16
2.25	0	1.55E+17	1.62E+17	1.10E+18	1.40E+18	1.48E+18	8.70E+17	2.16E+17	1.80E+16	2.27E+15	3.61E+14
2.75	0	5.31E+16	4.88E+16	2.15E+17	2.61E+17	2.40E+17	9.59E+16	1.39E+16	2.46E+15	4.43E+15	1.38E+15
3.5	0	1.90E+16	1.00E+16	2.92E+16	3.59E+16	3.37E+16	1.48E+16	2.82E+15	1.10E+14	4.62E+13	1.46E+13
5	0	5.97E+15	7.58E+14	3.67E+15	5.69E+15	6.95E+15	4.77E+15	1.09E+15	3.57E+13	1.52E+06	1.47E+06
7	0	0.00E+00	0.00E+00	6.16E+04	1.23E+05	2.46E+05	2.47E+05	2.47E+05	2.46E+05	2.44E+05	2.37E+05
9.5	0	0.00E+00	0.00E+00	9.62E+03	1.92E+04	3.85E+04	3.85E+04	3.85E+04	3.85E+04	3.82E+04	3.70E+04

Note:

1. A multiplier of 1.06 has been applied to the ORIGEN2 output to account for fuel management variations.

Table 7.6-11 Direct and Skyshine Control Room Dose and Dose Rate for the Alternate Source Term		
Time (hours)	**Dose Rate (rem/hr)**	**Dose (rem)**
0.167	3.88E-05	-
0.667	4.97E-05	6.08E-05
0.992	2.16E-04	9.75E-05
1.317	3.06E-04	1.82E-04
1.967	4.08E-04	4.12E-04
3.97	2.99E-04	1.11E-03
5	2.71E-04	1.41E-03
10	2.06E-04	2.59E-03
11.97	1.92E-04	2.98E-03
24	1.47E-04	5.01E-03
64	9.86E-05	9.86E-03
72	9.38E-05	1.06E-02
100	8.04E-05	1.31E-02
168	5.83E-05	1.77E-02
720	1.19E-05	3.39E-02
8766	5.80E-06	1.02E-01

Table 7.6-12

Parameters Used to Calculate Normal Operation Sources per ANSI/ANS-18.1-1984

Parameter	Symbol	Units	Kewaunee Value	ANSI Standard Nominal Value
Thermal power	P	MWt	1772	3400
Steam flow rate	FS	lb/hr	7.74E+06	1.5E+07
Weight of water in RCS	WP	lb	2.63E+05	5.5E+05
Weight of water in all steam generators	WS	lb	1.9E+05	4.5E+05
Reactor coolant letdown flow (purification)	FD	lb/hr	2.0E+04	3.7E+04
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	3.17E+01	5.0E+02
Steam generator blowdown flow (total)	FBD	lb/hr	3.3E+04	7.5E+04
Fraction of radioactivity in blowdown stream that is not returned to the secondary coolant system	NBD	--	1.0	1.0
Flow through the purification system cation demineralizer	FA	lb/hr	2.0E+03	3.7E+03
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	--	0.0	0.0
Fraction of the noble gas activity in the letdown stream that is not returned to the RCS	Y	--	--	0.0

Table 7.6-13 Normal Sources					
Group I – Noble Gases – No VCT Purge					
Nuclide	VCT Stripping Fraction	Reactor Coolant Activity (UCI/gram)	VCT Vapor Activity (UCI/CC)	GDT Inventory (Curies)	Steam Generator Steam Activity (UCI/gram)
KR-85M	6.0E-01	1.4E-01	1.3E+00	0.0E+00	5.6E-08
KR-85	7.2E-05	3.2E+00	7.4E+01	0.0E+00	1.3E-06
KR-87	8.4E-01	1.5E-01	5.4E-01	0.0E+00	5.7E-08
KR-88	7.0E-01	2.5E-01	1.7E+00	0.0E+00	1.0E-07
XE-131M	1.8E-02	6.9E-01	1.2E+01	0.0E+00	2.7E-07
XE-133M	8.9E-02	5.4E-02	8.5E-01	0.0E+00	2.2E-08
XE-133	3.9E-02	2.1E+00	3.5E+01	0.0E+00	8.6E-07
XE-135M	9.5E-01	1.4E-01	1.1E-01	0.0E+00	5.6E-08
XE-135	3.6E-01	6.9E-01	7.7E+00	0.0E+00	2.9E-07
XE-137	9.9E-01	3.7E-02	7.9E-03	0.0E+00	1.5E-08
XE-138	9.6E-01	1.3E-01	9.8E-02	0.0E+00	5.2E-08

Table 7.6-13 (Cont.) Normal Sources					
Group I -- Noble Gases -- 1.324E+02 CC/sec VCT Purge					
Nuclide	VCT Stripping Fraction	Reactor Coolant Activity (UCI/gram)	VCT Vapor Activity (UCI/CC)	GDT Inventory (Curies)	Steam Generator Steam Activity (UCI/gram)
KR-85M	7.3E-01	1.3E-01	8.0E-01	2.5E+00	5.4E-08
KR-85	5.5E-01	1.0E-02	1.1E-01	(1)	4.1E-09
KR-87	8.7E-01	1.5E-01	4.5E-01	3.9E-01	5.7E-08
KR-88	7.8E-01	2.5E-01	1.2E+00	2.4E+00	1.0E-07
XE-131M	4.8E-01	6.8E-02	6.1E-01	1.2E+02	2.7E-08
XE-133M	5.0E-01	2.1E-02	1.8E-01	6.5E+00	8.8E-09
XE-133	4.9E-01	4.3E-01	3.8E+00	3.3E+02	1.7E-07
XE-135M	9.5E-01	1.4E-01	1.1E-01	1.9E-02	5.6E-08
XE-135	6.0E-01	5.9E-01	4.1E+00	2.6E+01	2.4E-07
XE-137	9.9E-01	3.7E-02	7.8E-03	3.5E-04	1.5E-08
XE-138	9.6E-01	1.3E-01	9.4E-02	1.5E-02	5.2E-08

Table 7.6-13 (Cont.) Normal Sources	
Group I -- Noble Gases -- KR-85 Buildup	
Equivalent Full-Power Operation (Year)	GDT Inventory (Curies)
1	2.1E+02
2	4.0E+02
3	5.9E+02
4	7.6E+02
5	9.2E+02
6	1.1E+03
8	1.3E+03
10	1.6E+03
12	1.8E+03
15	2.1E+03
20	2.4E+03
25	2.7E+03
30	2.8E+03
35	3.0E+03
40	3.1E+03

Table 7.6-13 (Cont.) Normal Sources			
Nuclide	Reactor Coolant Activity (UCI/gram)	Steam Generator Liquid Activity (UCI/gram)	Steam Generator Steam Activity (UCI/gram)
Group II -- Halogens			
BR-84	1.7E-02	1.9E-07	1.9E-09
I-131	4.5E-02	4.1E-06	4.1E-08
I-132	2.3E-01	7.7E-06	7.7E-08
I-133	1.4E-01	1.1E-05	1.1E-07
I-134	3.7E-01	6.0E-06	6.0E-08
I-135	2.7E-01	1.6E-05	1.6E-07
Group III -- RB,CS			
RB-88	2.1E-01	1.3E-06	6.5E-09
CS-134	6.8E-03	6.4E-07	3.3E-09
CS-136	8.4E-04	7.8E-08	3.9E-10
CS-137	9.0E-03	8.5E-07	4.2E-09
Group IV -- N-16			
N-16	4.0E+01	2.3E-06	2.3E-07
Group V -- Tritium			
H-3	1.0E+00	1.0E-03	1.0E-03

Table 7.6-13 (Cont.)

Normal Sources

Group VI -- Other Isotopes							
Nuclide	Reactor Coolant Activity (UCI/gram)	Steam Generator Liquid Activity (UCI/gram)	Steam Generator Steam Activity (UCI/gram)	Nuclide	Reactor Coolant Activity (UCI/gram)	Steam Generator Liquid Activity (UCI/gram)	Steam Generator Steam Activity (UCI/gram)
NA-24	4.7E-02	3.4E-06	1.7E-08	RU-106	8.6E-02	8.0E-06	3.9E-08
CR-51	3.0E-03	2.8E-07	1.4E-09	RH-103M	7.2E-03	6.7E-07	3.5E-09
MN-54	1.5E-03	1.4E-07	7.1E-10	RH-106	8.6E-02	8.0E-06	3.9E-08
FE-55	1.1E-03	1.1E-07	5.4E-10	AG-110M	1.2E-03	1.1E-07	5.8E-10
FE-59	2.9E-04	2.6E-08	1.3E-10	TE-129M	1.8E-04	1.7E-08	8.4E-11
CO-58	4.4E-03	4.1E-07	2.0E-09	TE-129	2.6E-02	5.5E-07	2.7E-09
CO-60	5.1E-04	4.8E-08	2.4E-10	TE-131M	1.5E-03	1.2E-07	6.0E-10
ZN-65	4.9E-04	4.5E-08	2.2E-10	TE-131	8.4E-03	7.3E-08	3.8E-10
SR-89	1.3E-04	1.2E-08	6.3E-11	TE-132	1.7E-03	1.4E-07	7.2E-10
SR-90	1.1E-05	1.1E-09	5.4E-12	BA-137M	8.5E-03	8.0E-07	4.0E-09
SR-91	9.8E-04	6.5E-08	3.3E-10	BA-140	1.2E-02	1.1E-06	5.6E-09
Y-90	1.4E-06	1.3E-10	6.6E-13	LA-140	2.5E-02	2.1E-06	1.0E-08
Y-91M	5.0E-04	8.0E-09	4.0E-11	CE-141	1.4E-04	1.3E-08	6.7E-11
Y-91	5.0E-06	4.5E-10	2.4E-12	CE-143	2.8E-03	2.2E-07	1.1E-09
Y-93	4.3E-03	2.8E-07	1.4E-09	CE-144	3.8E-03	3.5E-07	1.8E-09
ZR-95	3.7E-04	3.5E-08	1.7E-10	PR-143	3.4E-03	2.6E-07	1.4E-09
NB-95	2.7E-04	2.4E-08	1.2E-10	PR-144	3.8E-03	3.5E-07	1.8E-09
MO-99	6.2E-03	5.5E-07	2.6E-09	W-187	2.5E-03	2.0E-07	9.9E-10
TC-99M	4.9E-03	2.6E-07	1.4E-09	NP-239	2.1E-03	1.9E-07	9.3E-10
RU-103	7.2E-03	6.7E-07	3.5E-09				

Table 7.6-13 (Cont.) Normal Sources			
CVCS Demineralizer Activity Concentrations			
Nuclide	Mixed Bed Activity (UCI/gram)	Nuclide	Cation Bed Activity (UCI/gram)
BR-84	1.4E-01	RB-88	1.0E+00
I-131	1.3E+02	CS-134	6.0E+01
I-132	7.8E+00	CS-136	4.5E+00
I-133	4.5E+01	CS-137	9.3E+01
I-134	4.9E+00	SR-89	1.1E-01
I-135	2.7E+01	SR-90	4.7E-03
RB-88	4.7E-01	SR-91	6.4E-03
CS-134	2.7E+02	BA-137M	8.8E+01
CS-136	2.0E+00	BA-140	2.6E+00
CS-137	4.1E+02		
NA-24	1.1E+01		
CR-51	3.0E+01		
MN-54	9.5E+01		
FE-55	9.2E+01		
FE-59	4.6E+00		
CO-58	1.1E+02		
CO-60	4.3E+01		
ZN-65	2.8E+01		
SR-89	2.4E+00		
SR-90	1.0E+00		
SR-91	1.4E-01		
BA-137M	3.9E+02		
BA-140	5.7E+01		

Table 7.6-14

Kewaunee Core Decay Heat after Shutdown
(core power = 1782.6 MWt)

Time after Shutdown (hours)	Total Decay Heat (watts)
0	1.15E+08
1	2.58E+07
2	2.09E+07
4	1.73E+07
8	1.45E+07
12	1.31E+07
16	1.21E+07
20	1.14E+07
24	1.08E+07
28	1.04E+07
32	1.00E+07
36	9.65E+06
40	9.34E+06
44	9.06E+06
48	8.80E+06
72	7.68E+06
96	6.91E+06
120	6.33E+06
144	5.87E+06
288	4.34E+06
720	2.79E+06

Contributions to decay heat include:

- Fission products
- Actinides
- Assembly activation products
- Reactor internals activation products
- Fuel management variations

Table 7.6-15

Kewaunee High-Burnup Assembly - Decay Heat after Shutdown at 62,000 MWD/MTU Assembly Burnup (enrichment = 4.5 w/o, assembly power = 14.73 MWt)

Time after Shutdown	Time Units	Decay Heat (watts)
0	d	8.80E+05
10	d	3.93E+04
30	d	2.47E+04
60	d	1.81E+04
90	d	1.51E+04
120	d	1.31E+04
180	d	1.04E+04
1	yr	6.63E+03
2	yr	3.90E+03
5	yr	1.71E+03
10	yr	1.11E+03
20	yr	8.17E+02
30	yr	6.60E+02
40	yr	5.46E+02
50	yr	4.59E+02
60	yr	3.91E+02

Contributions to decay heat include:

- Fission products
- Actinides
- Assembly activation products

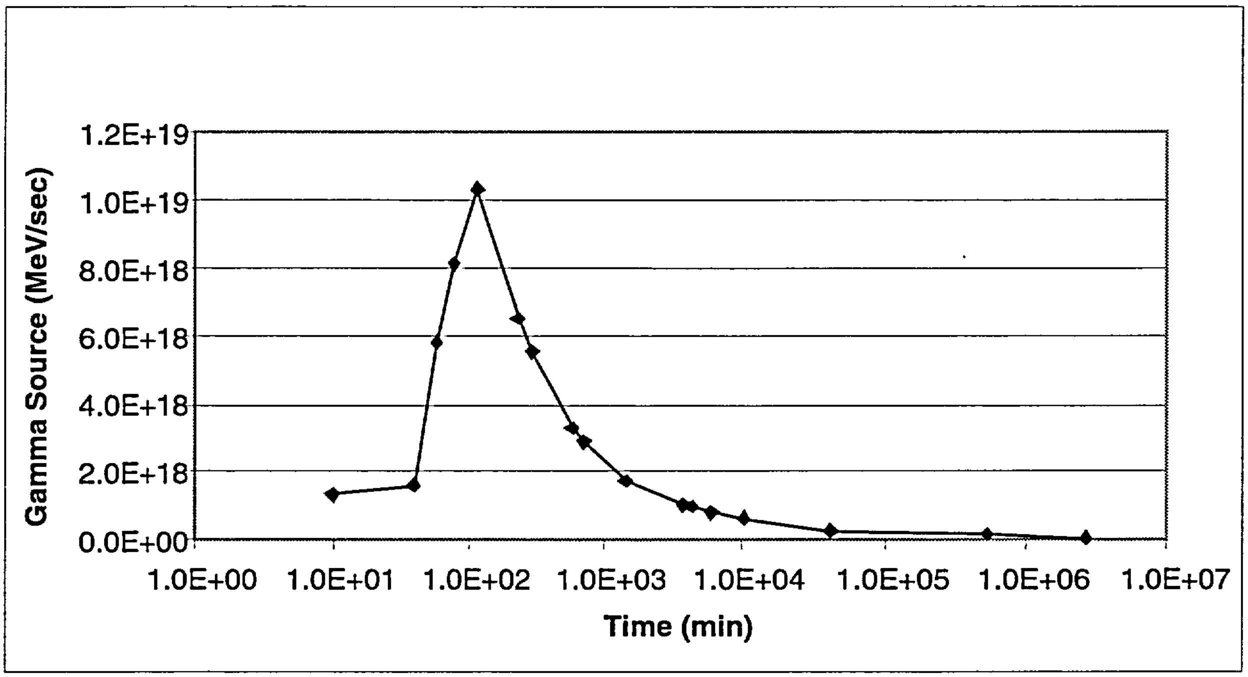


Figure 7.6-1

DBA Gamma Source versus Time - MeV/sec

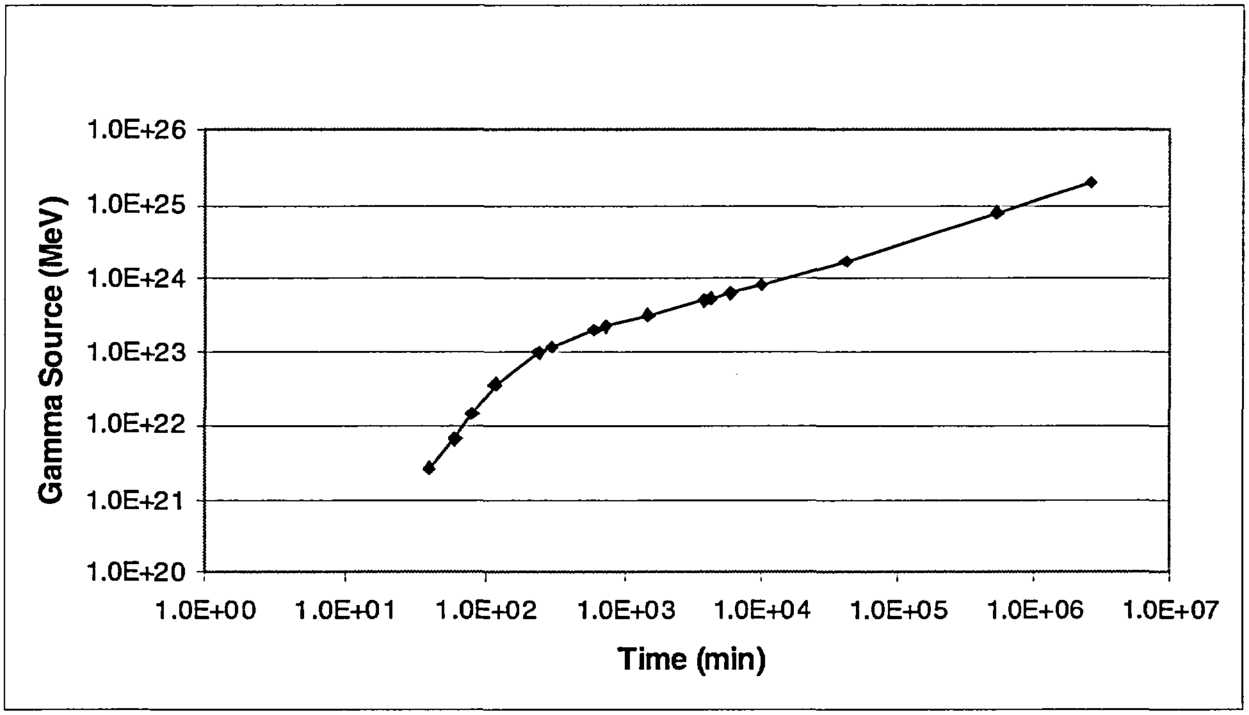


Figure 7.6-2

Integrated Gamma Source versus Time - MeV

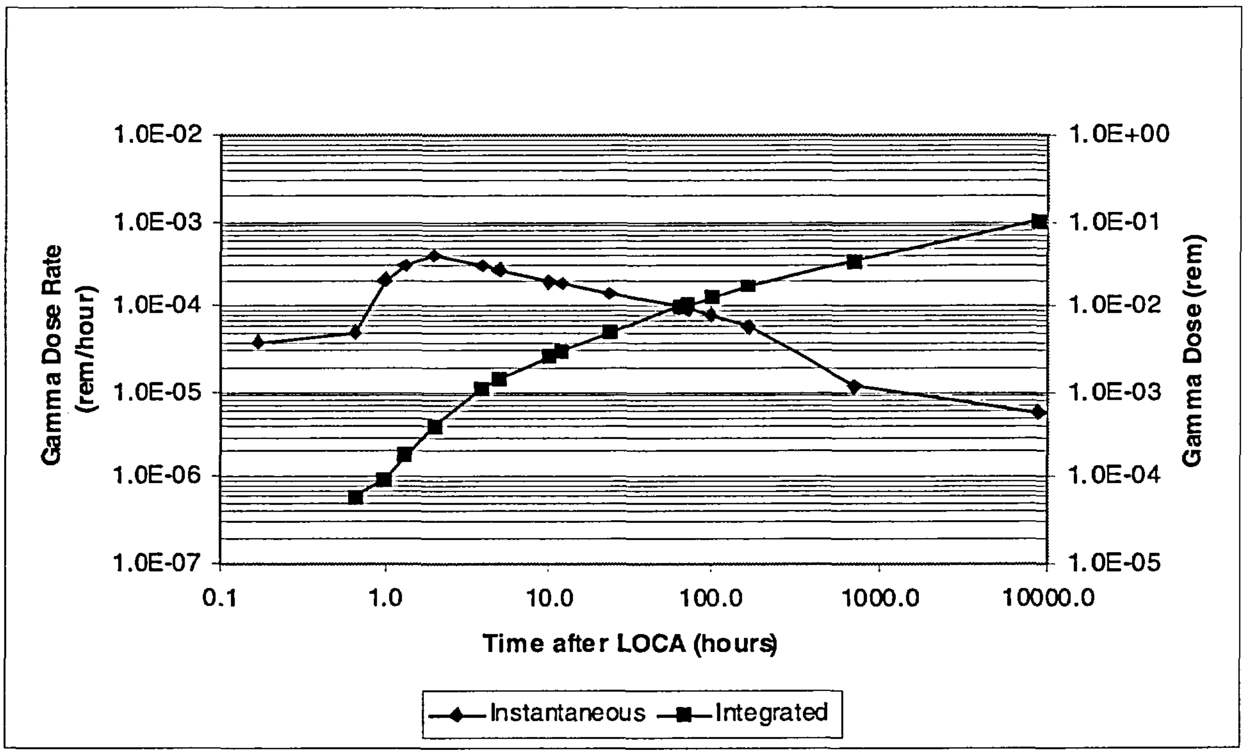


Figure 7.6-3

Gamma Dose in Control Room after LOCA DBA

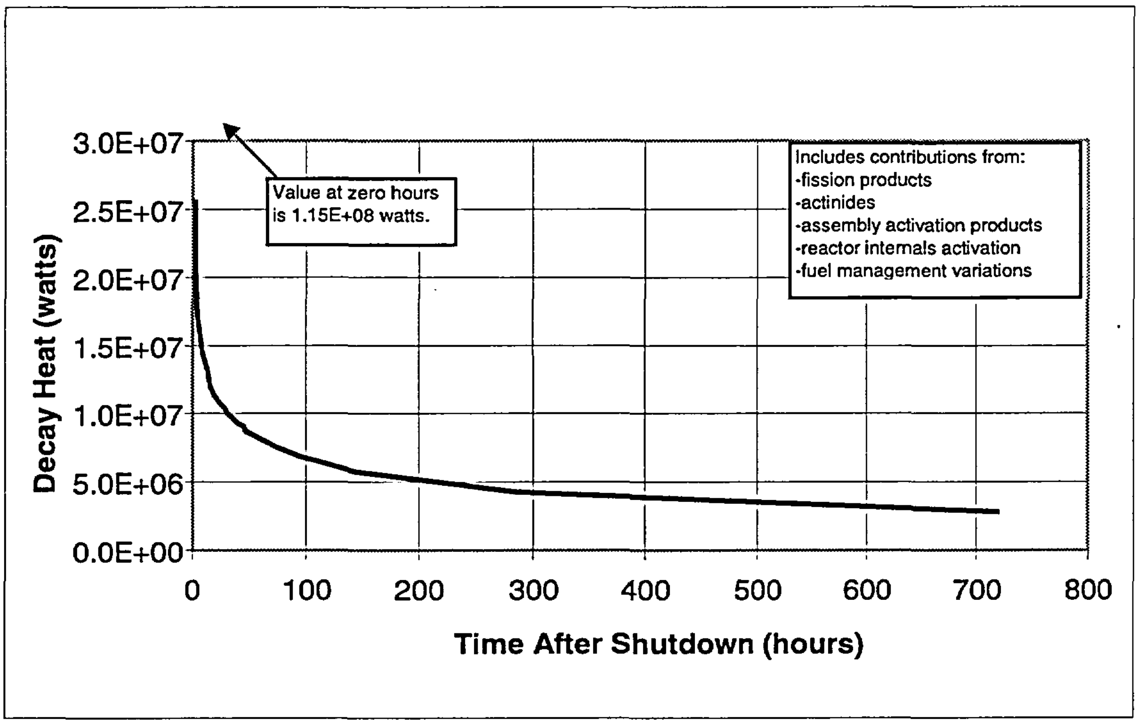
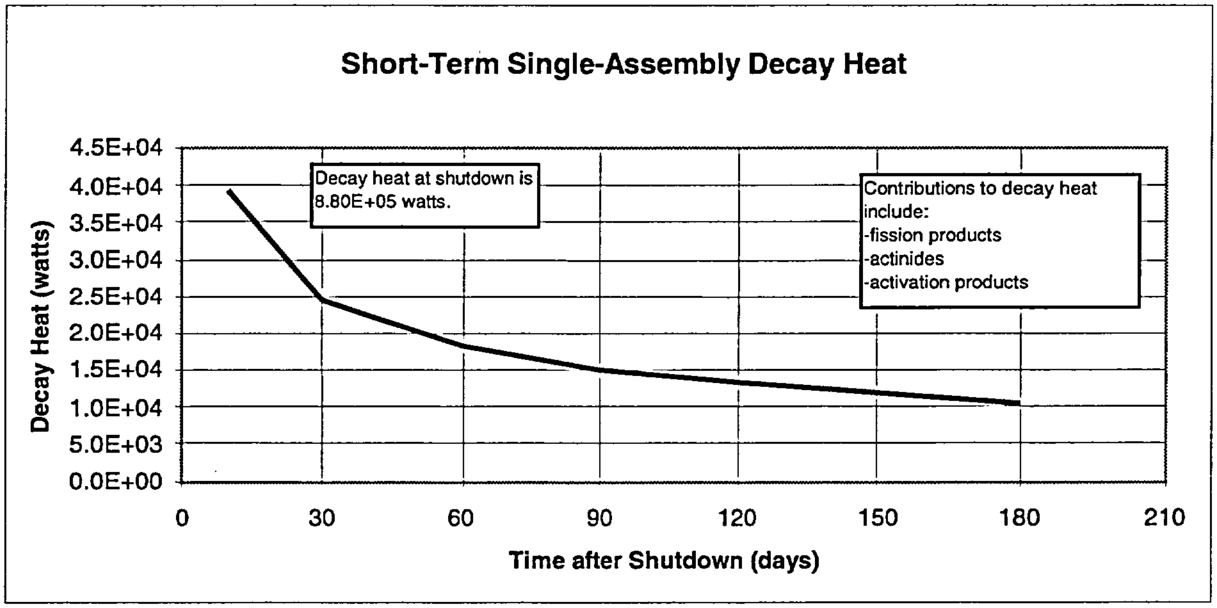


Figure 7.6-4

Kewaunee Core Decay Heat for the Uprate

(core power = 1772 * 1.006 = 1782.6 MWt)



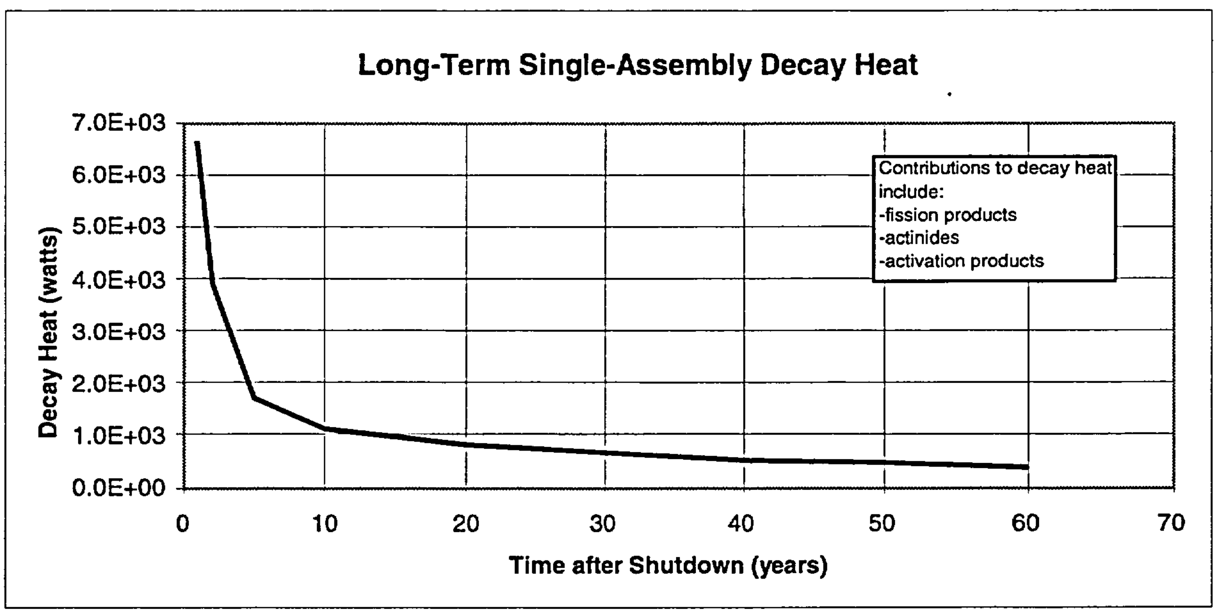


Figure 7.6-5

Short-Term and Long-Term Single-Assembly Decay Heat - 62,000 MWD/MTU Burnup, 4.5 w/o enrichment

8.0 BALANCE OF PLANT

This section of the Licensing Report documents the evaluations conducted to assess the balance-of-plant (BOP) structures, systems, and components to ensure that they are structurally and functionally capable of safe, reliable operation at the power uprate conditions. The study included a review of major BOP components and systems impacted by the power uprate.

The BOP engineering and associated reviews, evaluations, calculations, and analyses required to support the Kewaunee Power Uprate Project were performed at the power uprated Nuclear Steam Supply System (NSSS) power level of 1780 MWt.

Current and Power Uprate Power Levels		
Description	**Core Power (MWt)**	**NSSS Power (MWt)**
Current Licensed Power Level	1650	1656
Uprate Power Levels (7.4-percent power uprate)	1772	1780

The evaluations were performed based on the existing design and licensing basis documented in the *Updated Safety Analysis Report* (USAR) and Technical Specification bases. When either the existing basis could not be met following power uprate, or a revised basis was used to demonstrate compliance to new criteria, justification for compliance and/or the revised basis is provided in the acceptance criteria used for the power uprate evaluation. In addition, calculations were performed in areas where existing documentation did not demonstrate capability at the power uprate conditions.

Results of the detailed BOP evaluations demonstrate that Kewaunee is capable of providing safe and reliable operation at the Nuclear Steam Supply System (NSSS) power level of 1780 MWt with no major modifications.

Only three minor hardware modifications were needed:

- Modified valve trim in the feedwater control valve (FRV)
- High-pressure turbine horizontal bolt ring changes
- Low-pressure turbine coupling bolts changes

BOP instrument scaling and setpoint changes required to support the power uprate are as follows:

- Condensate storage tank (CST) Technical Specification level
- Turbine overspeed setpoints

Evaluations were performed utilizing heat balances (discussed below) and the NSSS parameters as described in Section 2 of this Licensing Report.